

International Agreement Report

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

Prepared by: Chunkuan Shih*, Hao-Chun Chang*, Jong-Rong Wang*, Shao-Wen Chen*, Hao-Tzu Lin, Show-Chyuan Chiang**, Tzu-Yao Yu**

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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.

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

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.

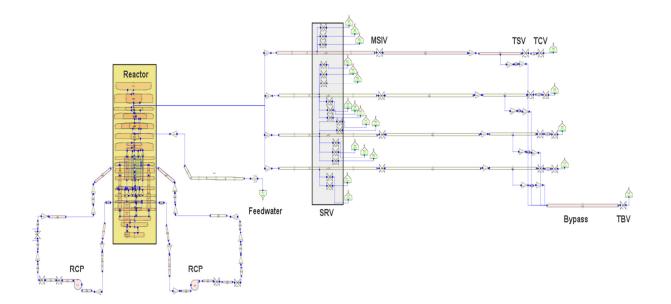


Figure 1 Overview of the Kuosheng NPP RELAP5/MOD3.3 Model

Reactor Kinetics

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.

*	Reactor Kinetics	X	<u>à</u> '	2
▼ General	Show Disa	able	d	?
Description	<none></none>	E₹	2	?
Enable	🖲 True 🔾 False		2	?
Kinetics Type	Point		2	?
Feedback Type	SEPARABL		2	?
Fission Product Decay	GAMMA-AC	•	2	?
Total Reactor Power	2.757e9 (W)	٩Þ	2	?
Initial Reactivity	0.0 (\$)	٩٥	2	?
Beta Over Lambda	125.0 (s ⁻¹)	٩٥	2	?
Product Yield Factor	✓ 1.0 (-)	٩٥	2	?
U239 Yield Factor	✓ 0.7 (-)	٩٥	2	?
Fissions Per Atom	✓ 1.0 (-)	٩⊳	2	?
Reactor Time	✓ 1.2614e8 (sec))	2	?
Differential Boron Worth	Unknown (-)	${\triangleleft}{\triangleright}$	2	?
Boron Feedback	<pre>Inactive ></pre>	-	2	?
Delayed Neutron Const.	Rows: 0 []	E₹	2	?
Fission Product Decay	Rows: 0 []	E٩	2	?
Actinide Decay	Rows: 0 []	E٩	2	?
Power History Table	[0] Power History Rows	E₹	2	?
Reactivity Control	XIY [1]General Table 900	S₹	2	?
Density Reactivity	Rows: 11 [1.0,-10.17],[2.51115,-9.8	E₹	2	?
Doppler Reactivity	Rows: 5 [400.0,4.295],[800.0,2.148]	E₹	2	?
Volume Weighting	[3] Volumes	E₹	2	?
Heat Weighting	[6] Heatstructures	E₹	2	?
Model Detector Data	🔾 True 💿 False		2	?
▼ [3000002] Fission P	roduct Decay			
Enable 🔾 True 🖲 Fal	se		2	?
▶ [30000004] Nodal Kin	etics Control Parameters			
► [30000010] Krylov Sol	ution Algorithm			

Figure 2 Parameters of Reactor Kinetics in RELAP5 Model

🔇 Editing Density Reactiv	ity X
Moderator Density	Reactivity
lbm/ft ³	\$
1.0	-10.17
2.51115	-9.85
10.511155	-7.17
14.511378	-5.12
17.511489	-2.06
18.5116	-0.1
31.11614	0.0
35.88319	2.26
40.65024	4.57
45.41729	6.87
50.18434	9.17
	Add Remove
ОК	Cancel



C	Editing Doppler React	iivity 📃	x
	Temperature	Reactivity	
	F	\$	
	400.0	4.295	•
	800.0	2.148	
	1200.0	0.0	=
	1600.0	-2.148	
	2000.0	-4.295	-
		Add Remove	
	ОК	Cancel	

Figure 4 Doppler Reactivity Table of Reactor Kinetics

🔇 General Table 900	- Properties View	23	\$
— XIY General Table	900		Ĩ
▼ General	Show D	isabled	1
Name	unnamed	1 🔁 🤋	?
Component Number	900	9 🕾 💈	?
Description	<none></none>	• 🕾 💈	?
Table Type	Reactivity Vs Time	• 🕾 💈	?
Table Trip	Variable Trip 406	• 🕾 💈	?
Independent Factor	1.0 (-) 4	2	?
Dependent Factor	1.0 (-) 4	2	?
Table Data	Rows: 18 [0.0,0.0],[0.15,-0.27272727],[0.45,-0 E	• 🕾 🖇	?
Editing Table Data		×	3

O Editing Table Data

Shared Table None Selected	ST
Time	Reactivity
s	\$
0.0	0.0
0.15	-0.27272727
0.45	-0.81818182
0.6	-4.0
0.7	-6.0
0.8	-7.0
1.0	-8.0
1.36842	-10.5
1.73684	-11.55
2.13158	-14.85
2.97368	-20.0
3.78947	-30.0
4.63158	-30.0
5.21053	-30.0
5.84211	-30.0
6.89474	-30.0
8.0	-30.0
9.0	-30.0
ОК	Sort Add Remove

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.

						_
onnections						
MSIV						2 3
er					542	2 3
<none></none>					E	2 3
28 PIPE 541						23
C: PIPE 543						2 3
Cells: 0 / Juns.: 1					E٦	23
Valid values					E٦	23
Valid values					E٦	23
Motor Valve					-	23
⊳ Logical Trip 619	(MSIV)				S	2 3
⊳ Variable Trip 531	I				S	23
				0.205 (-) 🕸	23
				1.0 (-) 🕸	23
General Table 1	1 (MSIV tabl	e)			S₹	2 3
te 🔲				known (-) 💿	2
🔾 True 💿 False						2 3
Rows: 0 []					E٦	23
VALVE 542 (MSIV)						X
Hydraulic Use Mod	. Choking	Single Vel	Area	Momen	tun	Area
		Momentun	Change			(ft ²) 2.53
ditional Junction Data						
	Close					
ction Data - VALVE	542 (MS	IV)				X
Forward		Reverse		I	Use	
Loss	15	Loss	4.5	Re	ynolds	3
	1.5		1.5			
	MSIV per <none> Cells: 0 / Juns.: 1 Valid values Valid values Valid values Motor Valve >- Logical Trip 619 >- Variable Trip 531 Image: Comparison of the system Quid values Valid values Motor Valve >- Logical Trip 619 >- Variable Trip 531 Image: Comparison of the system Quid value Prove (Comparison of the system) Image: Comparison of the system <t< td=""><td>Image: Second constraints Image: Sec</td><td>Image: Second constraints Image: Sec</td><td>MSIV per <none> % PIPE 541 % PIPE 543 Cells: 0 / Juns.: 1 Valid values Valid values Valid values Motor Valve ▷• Logical Trip 619 (MSIV) ▷• Variable Trip 531 ☑ ☑ General Table 11 (MSIV table) te ☑ ☑ Ørage Prue @ False Rows: 0 [] VALVE 542 (MSIV) Area Model Momentun Change 0.0 0 1 Iditional Junction Data Close Ction Data - VALVE 542 (MSIV)</none></td><td>Image: Second State Sta</td><td>Image: Street Stree</td></t<></none>	Image: Second constraints Image: Sec	Image: Second constraints Image: Sec	MSIV per <none> % PIPE 541 % PIPE 543 Cells: 0 / Juns.: 1 Valid values Valid values Valid values Motor Valve ▷• Logical Trip 619 (MSIV) ▷• Variable Trip 531 ☑ ☑ General Table 11 (MSIV table) te ☑ ☑ Ørage Prue @ False Rows: 0 [] VALVE 542 (MSIV) Area Model Momentun Change 0.0 0 1 Iditional Junction Data Close Ction Data - VALVE 542 (MSIV)</none>	Image: Second State Sta	Image: Street Stree

Figure 6 Properties of Main Steam Isolation Valves

VALVE 546 (1	SV) - Properties View	
🔶 🖫 Hydro Co	nnections	-
▼ General	Show	Disabled
Name	TSV	2
Component Numbe	r 54	16 🔁 🧣
Description	<none></none>	e 1 🔁 💈
Inlet	t; PIPE 545	2 3
Outlet	업 PIPE 547	2
Geometry	Cells: 0 / Juns.: 1	2 🕈
Friction Data	Valid values	2 🔁 💈
Initial Conditions	Valid values	2 🕈
Valve Type	Motor Valve	- 🕈 💈
Open Trip	⊷ Logical Trip 603 (TSV)	s 🕈 🕈
Close Trip	⊳ Variable Trip 407	57 🔁 💈
Change Rate	10.0 (-)	1> 🕾 🖇
Initial Position	1.0 (-)	1> 🕾 🖇
Valve Table	<pre>Inactive ></pre>	s 🕈 🕫
Close Change Rate	Unknown (-)	1> 🔁 💡
Table Factors	◯ True	2 3
Motor Servo Table	Rows: 0 []	ET 🔁 💈
🗿 Geometry - V	ALVE 546 (TSV)	X
Junction Use	Hydraulic Use Mod. Choking Single Vel Area Momentun	Area
Number CCFL	Diam. (ft) PV Term Model Momentun Change Flux 0.0 0 1 0	(ft ²) 2.536
Junctions Addi	ional Junction Data	
	Close	
Junction Fric	ion Data - VALVE 546 (TSV)	X
Junction	Forward Reverse Use	
Number 1	Loss Loss Reyno 1.5 1.5	ds
1	6.1 6.1	
Junction Friction		
	Close	

Figure 7 Properties of Turbine Stop Valves

ዮ─ ₩ VALVE 548 ፦ ℃ Hydro			
▼ General		Show Dis	
Name	TCV		2
Component Num	ber	548	2
Description	<none></none>	E	2
Inlet	C PIPE 547		?
Outlet	C TMDPVOL 549 (MSL1SRVS)		2
Geometry	Cells: 0 / Juns.: 1	E٦	2
Friction Data	Valid values	E٦	2
Initial Conditions	Valid values	E٦	2
Valve Type	Motor Valve	-	2
Open Trip	Logical Trip 603 (TSV)	S	?
Close Trip	≫ Variable Trip 407	S	?
Change Rate		10.0 (-) 🕸	<u>9</u>
Initial Position		1.0 (-) 40	e
Valve Table	<pre>Inactive ></pre>	S	P
Close Change R	ite 🔲 Unkr	10wn (-) ∢⊳	2
Table Factors	🔾 True 🖲 False		<u>e</u>
Motor Servo Tabl	Rows: 0 []	E٦	2
Geometry -	VALVE 548 (TCV)		X
Junction Use Number CCF			Area (ft²)
1	Diam. (ft) PV Term Model Momentum Change 0.0 0 1	0	2.53
Junctions	ditional Junction Data		
Junction Fr	Close ction Data - VALVE 548 (TCV)		X
Junction	Forward Reverse	Use	
Number	Loss Loss	Reynolds	
1	0.0 0.0		
Junction Frictio			

Figure 8 Properties of Turbine Control Valves

- 🚖 TMDPVOL 539	(MSI 1SRVS)						
P 🙆 TMDPVOL 539				-			
✓ General			Show Dis	abled			
Vame	MSL1SRVS		SHOW DIS	eneu ?			
Component Number			539	39 🔁 🥐			
Description	<none></none>		E٦	۳ ?			
Geometry	Cells: 1 / Juns.: 0		E	2 ?			
Fluid Conditions	Valid values		E	۳ 🕈			
Frip	<pre>< Inactive ></pre>		S	۲ 🕈			
/ariable Request	<pre>< Inactive ></pre>		S	۲ ?			
Noncondensable	Gas Species						
Metal Species							
				X			
Geometry - T	MDPVOL 539 (MSL1S	RVS)					
Cell	Volume	Length	Area				
Number	(ft ³)	(ft)	(ft²)				
Total	1.0e6			1.0e4			
	Calculate: 🔘 Volum	e 🖲 Length 🔾 Area					
Cells Orientation	1						
	Clo	ose					
Cell IC Data -	IMDPVOL 539 (MSL1	SRVS)		X			
	Search	Pressure	Static				
Table Entry							
Number	Variable	(psi)	Quality	10			
	Variable 0.0			1.0			
Number 1				1.0			
Number				1.0			
Number 1	B Flag Input	965.4 Table					
Number 1 T Flag Input Input Type 2	B Flag Input	965.4 Table	Data				
Number 1 T Flag Input Input Type 2	B Flag Input Discondensibles	965.4 Table	Data				

Figure 9 Time Dependent Volume Properties of the Main Steam Pipe Lines

O VALVE 441 (SRV-1) - Properties View						
୧− ତ4 VALVE 441 (Si ତି ଅଟି Hydro Con		•				
▼ General Show Disabled						
Name	SRV-1					
Component Number	441	2 2				
Description	<none> E</none>	2 2				
Inlet	ቲያ PIPE 541	2 2				
Outlet	C TMDPVOL 841 (MSL1SRVS)	2 ?				
Geometry	Cells: 0 / Juns.: 1	2 2				
Friction Data	Valid values E	• 🕾 🤋				
Initial Conditions	Valid values E	2 2				
Valve Type	Motor Valve	• 🔁 🎖				
Open Trip	🗫 Variable Trip 464 S	2 2				
Close Trip	🗫 Variable Trip 465 S	2 ?				
Change Rate	e 6.67 (-) 4 ▶					
Initial Position	0.0 (-) 4Þ					
Valve Table	S ^q					
Close Change Rate	Unknown (-) 🜗					
Table Factors	◯ True	2				
Motor Servo Table	Rows: 0 [] E	• 🕾 🨵				
📀 Geometry - V	ALVE 441 (SRV-1)	22				
Junction Use	Hydraulic Use Mod. Choking Single Vel Area Momentun	Area				
Number CCFL	Diam. (ft) PV Term Model Momentun Change Flux	(ft ²)				
	0.99032 0 1 0	0.19257				
O Junction IC Data - VALVE 441 (SRV-1)						
Velocities Mass Flow ?						
Junction	Junction Liquid Vapor					
Number	Velocity (ft/s) Velocity (ft/s)	0.0				
1	0.0	0.0				
Junction Flow						
	Close					
	Close					

Figure 10 Properties of Safety/Relief Valves

Pumps

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.

③ TMDPVOL 400 (FDWATV40) - Properties View ∑						
▼ General			Show Dis			
Name	FDWATV40			2 ?		
Component Number			400	2 ?		
Description	<none></none>		E	2 2		
Geometry	Cells: 1 / Juns.: 0		E	2 2		
Fluid Conditions	Valid values		E	2 2		
Trip	<pre>< Inactive ></pre>		S	2 ?		
Variable Request	<pre>< Inactive ></pre>		ST	2		
Noncondensable	Gas Species					
 Metal Species 			_	_		
📀 Cell IC Data -	TMDPVOL 400	(FDWATV40)		X		
Table Entry	Search	Pressure	Temp			
Number	Variable 0.0	(psi) 1065.0	(F)			
T Flag Input	B Flag Input	Table Data				
Input Type 3	Include Be	oron Add	Rem	ove		
TDV Fluid TDV N	oncondensibles	20				

Figure 11 Details of Feedwater Time Dependent Volume

▼ General	Show Dis		
Name	FDWATJ40	2	8
Component Number	401	2	8
Description	<none> E[*]</none>	2	8
Inlet	C을 TMDPVOL 400 (FDWATV40)	2	8
Outlet	다. PIPE 402 (FDWATV40)	2	8
Flow Area	0.94533333 (ft²) 🕸	2	8
Use PV Term	◯ True	2	?
Control Word	🖌 [1] Use Massflows 🗸	2	?
Trip	✓ <none> S</none>	2	?
Variable	✓ CB 994 sum (FW-24)	2	?
Table	Rows: 3 [-1.0e50,0.0,0.0],[0.0,0.0,0.0],[1.0e4,1 E	2	?

Figure 12 Details of Time Dependent Junction of Feedwater Pumps

PUMP 310 (0)	CPUM	PA31) - Pro	perties Vie	N			X	ζ
ዮ 🕂 PUMP 310 (A31)						7
⊶ 🚼 Hydro C ⊶ ቻ Heatstru								
▼ General					Sh	ow Dis	-	F
Name		JMPA31				_	* *	-
Component Numb	per						2 ?	
Description							2 ?	
Geometry	-	s: 1 / Juns.: 2					2 ?	
Friction Data		i values					° ?	
Initial Conditions		i values				E	2 ?	
▼ [CCC0301] Pur								
Pump Indicator		d with this com					*** ?	
Multiplier Index		nultiplier, ente					*** ?	
Difference Table		nase included		р			° ?	
Motor Torque		que Table Dat					* ?	
Pump Velocity		rque-inertia eq	luation				*** *	
Motor Trip		able Trip 405					** ? ** @	
Reverse Indicator						-	2 ?	
▼ [CCC0302-4] P	ump De	scription			4705.0 (<u>a</u> 0	41
Rated Velocity					1785.0 (rpr			
Ratio Velocity		1.0 (-)						
Rated Flow Rated Head		3.25e4 (gpm) 4					** ? ** ?	
Rated Torque								
Rated Density		1.7185e4 (lbf*ft) 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2						
Moment of inertia		53.481666 (Ibm/ft ³) 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2						
Rated Motor Torqu	ie				0.0 (lbf*		2 2	
TF2 Torque					0.0 (lbf*		2 2	
TF0 Torque					387.0 (lbf*		2 ?	
TF1 Torque					0.0 (lbf*	ft) ⊲⊳	2 ?	Π
TF3 Torque					0.0 (lbf*	ft) ⊲ ⊳	2 ?	1
▶ [CCC0308] Var	iable Ine	rtia						1
ICCC03091 Pur	no-Shafi	Connection						-
			Close					
Cell/Junction	n IC Da	ta - PUMP 3	310 (CPUM	PA31)			X	S
Cell	ndition	Use	Pressure	L.Energy	V.Energy	Vo	id	٦
Number		ndition Boron (psi) (btu/lbm) (btu/lbm)		Fra				
1 [P,U _f	^{'a'a}]		1177.607	521.309	1105.062		0.0	
					S	team T	ables	٦
Cell Fluid Nond	ondensi	bles Junct	ion Flow					
			Close					
			01030					

Figure 13 Recirculation Pumps Properties

VALVE 313 (RE)	CRAJ31) - Properties View	X					
P ♦ VALVE 313 (RECRAJ31) Image: Connections Image: Connections							
▼ General	Show D	-					
Name	RECRAJ31	2 2					
Component Number	313	3 🔁 🎖					
Description	<none></none>	• 🕾 😵					
Inlet	많 BRANCH 312 (RECRAV31)	2 ?					
Outlet	많 PIPE 314 (RECRAV31)	2 2					
Geometry	Cells: 0 / Juns.: 1						
Friction Data		• 🕾 😵					
Initial Conditions	Valid values E	• 🕾 😵					
Valve Type	Servo Valve 🔽 🕈 ?						
Control Variable	🗗 CB 617 mult S 🕈 🕾 🔋						
Valve Table	S	* 🕈 💡					
Table Factors	◯ True	2 ?					
Motor Servo Table	Rows: 0 [] E	• 🕾 😵					
Geometry - V.	ALVE 313 (RECRAJ31)	X					
Junction Use	Hydraulic Use Mod. Choking Single Vel Area Momentun	Area					
Number CCFL	Diam. (ft) PV Term Model Momentun Change Flux (ft ²						
1	0.0 0 1 0	1.7977					
Junctions Addit	ional Junction Data						
	Close						
	Close	_					
	01000						

Figure 14 Properties of Recirculation Controlling Valves

JETMIXER 340 (JETPAV34) - Properties View						X		
P- II JETMIXER 340 (JETPAV34) → C Hydro Connections → II: Heatstructure Connections								
▼ General						how Dis		
Name	JETPA	/34					2 ?	
Component Num	ber					340	2	
Description	<none></none>						2. 2.	
Geometry		/Juns.: 3					2 2	
Friction Data	Valid va						22	
Initial Conditions	Valid va						₽°.	
🛛 Geometry -	JETMIXE	R 340 (JET	PAV34)				X	
Cell		Volume		Length		Area		
Number		(ft ³)		(ft)		(ft ²)		
1 Total			85625 85625		.875 .875		1.9635	
Coll (hun atia		Data II	Close		(2.4)		57	
Cell/Junctic	on Friction	n Data - JE	TMIXER 34	40 (JETPAV	34)		X	
Cell		Use	e Non-Equilibi Model	rium	Com Frid			
1			₩00er		L L L	_		
Cell Options Wall Friction Junction Friction ORNL Data								
Cell/Junction IC Data - JETMIXER 340 (JETPAV34)								
Cell	ondition	tion					id	
Number	,U _g ,a _g]	Boron (psi) (btu/lbm) (btu/lbm) Frac.						
Steam Tables								
Cell Fluid Non	condensible	es Junct	ion Flow					
			Close					

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.

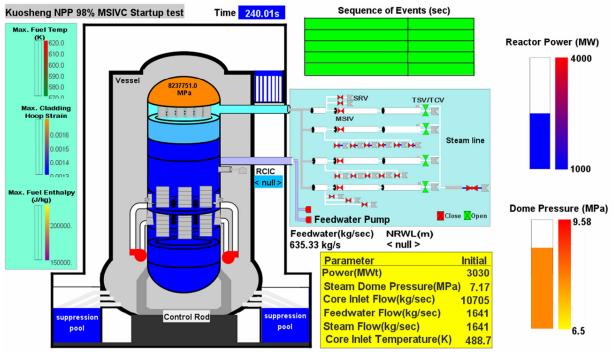


Figure 16 RELAP5/MOD3.3 Animation Model

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 <u>100% Load Rejection with Bypass Valve Open (LRWB)</u>

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.

e and etailup foot Bat	a III 100 /0 EITHE ITAIloiont
Test data	RELAP5
2894.0	2894.0
6.96	6.99
1551.66	1574.11
1510.91	1574.10
90	89.9
'81/11/11	_
	Test data 2894.0 6.96 1551.66 1510.91 90

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

 Table 2
 Event Sequence of 100% LRWB Transient
 Event(sec) Test data RELAP5 TCV Start to Close 210.2 0.2 Reactor Scram 0.236 210.29 **BPV Fully Open** 0.329 210.32 **TCV Fully Close** 0.394 210.35 Peak Dome Pressure Time 3.9 (7.43MPa) 212.84 (7.55MPa) (value) End of Analysis ____ 230

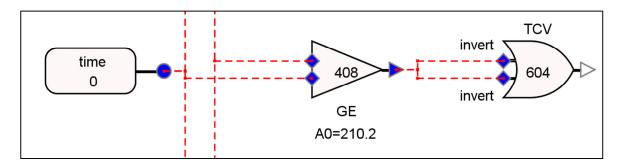


Figure 17 Controlling System of Turbine Control Valves in 100% LRWB Transient

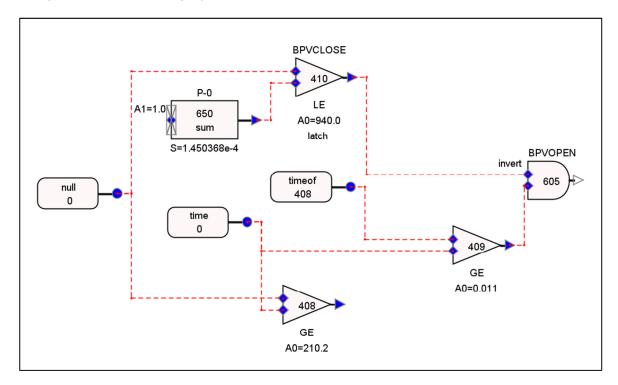


Figure 18 Controlling System of Turbine Bypass Valve in 100% LRWB Transient

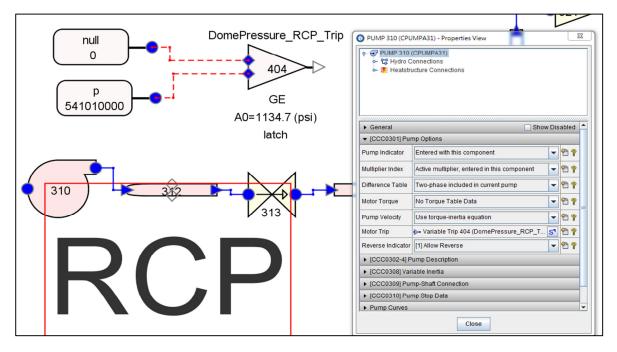
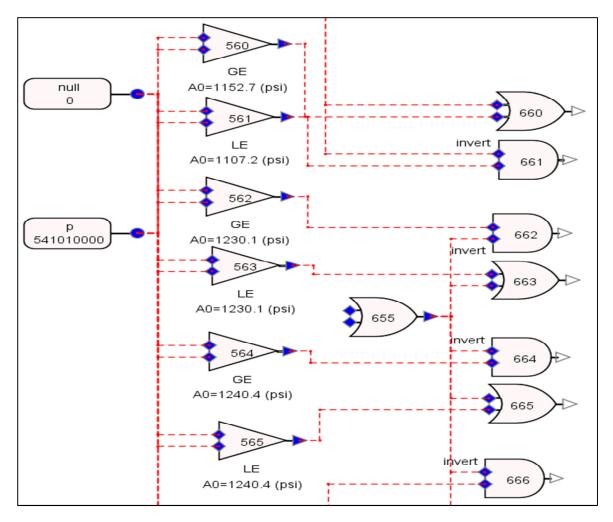


Figure 19 Controlling System of Recirculation Pumps Trip





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.

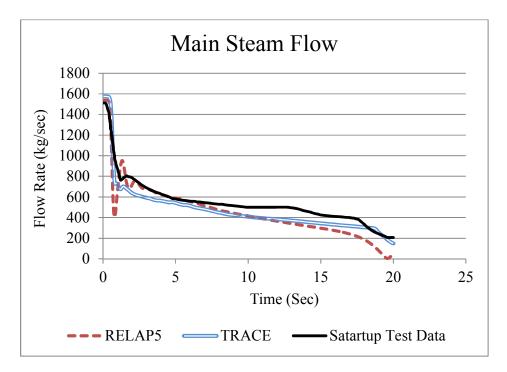


Figure 21 Steam Flow Variation during the 100% LRWB Transient

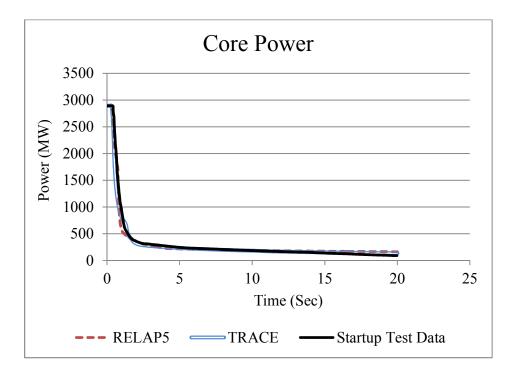


Figure 22 Core Power Variation during the 100% LRWB Transient

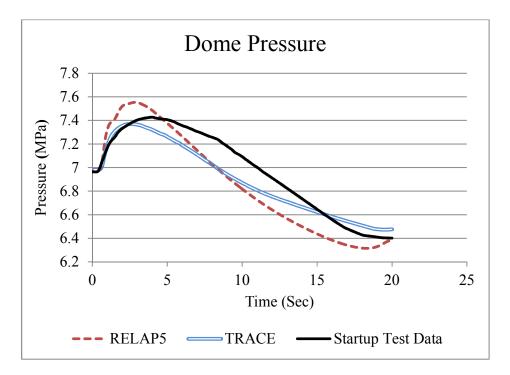


Figure 23 Dome Pressure Variation during the 100% LRWB Transient

3.2 <u>96% Main Steam Line Isolation Valves Closure (MSIVC)</u>

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.

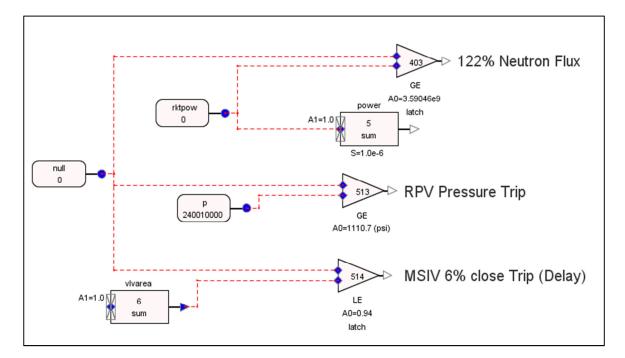
From the startup test results, the reactor scram as the MSIVs reached to 94% open. The MSIVs closure time was about 4.08 to 4.53 seconds. The dome pressure increment was about 118 psi during the transient. Because the dome pressure reached to the setpoints of the safety/relief valves, there were two 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.

	o una otartap rest bate	
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 3
 Initial Conditions of RELAP5 and Startup Test Data in 96% MSIVC Transient

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	_	240				





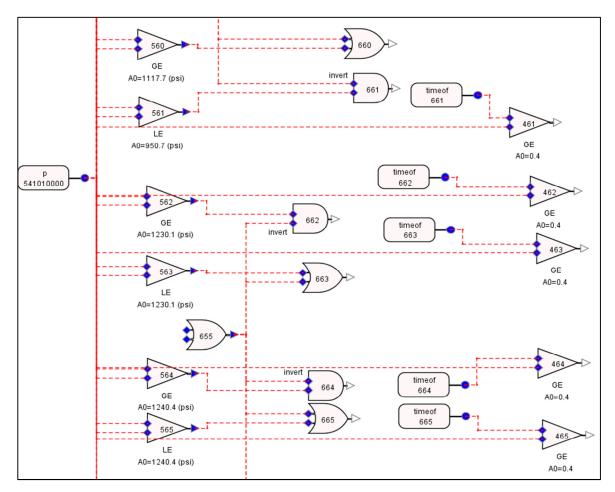


Figure 25 Controlling System of Safety/Relief Valves in 96% MSIVC Transient

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.

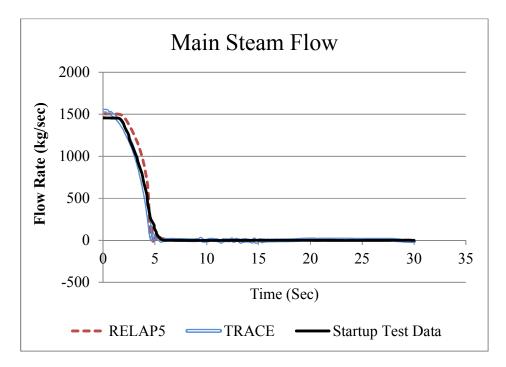


Figure 26 Steam Flow Variation during the 96% MSIVC Transient

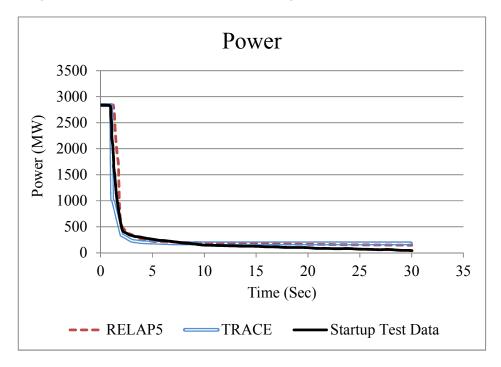


Figure 27 Core Power Variation during the 96% MSIVC Transient

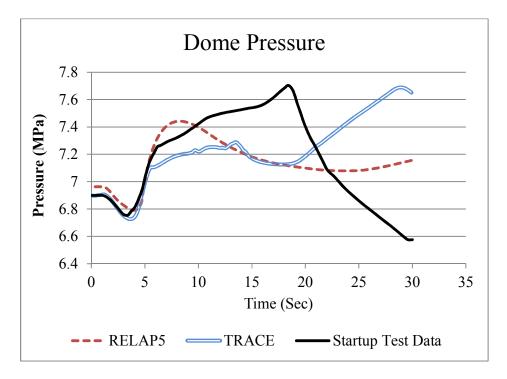


Figure 28 Dome Pressure Variation during the 96% MSIVC Transient

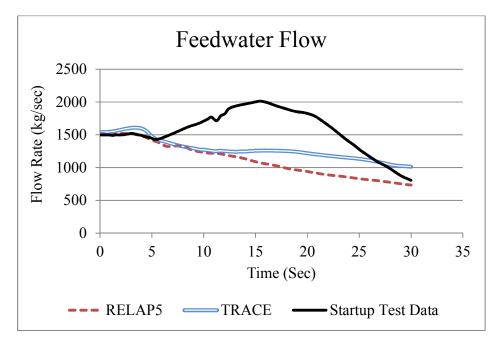


Figure 29 Feedwater Flow Variation during the 96% MSIVC Transient

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 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 RELA	P5 and Startup Test Data	a in 94% FWP1 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 5 Initial Conditions of RELAP5 and Startup Test Data in 94% FWPT Transient

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					

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Figure 30 Controlling System of Feedwater Pump Trip in 96% FWPT Transient

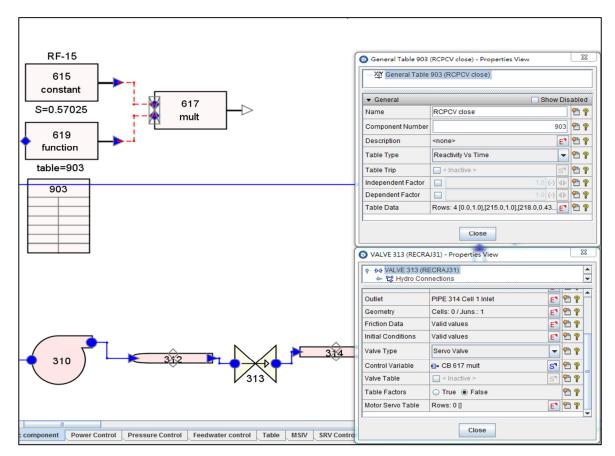


Figure 31 Controlling System of Flow Control Valves in 96% FWPT Transient

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.

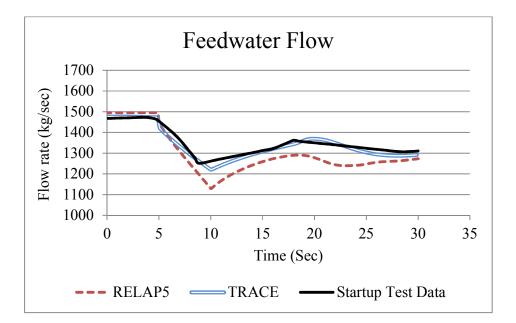


Figure 32 Feedwater Flow Variation during the 94% FWPT Transient

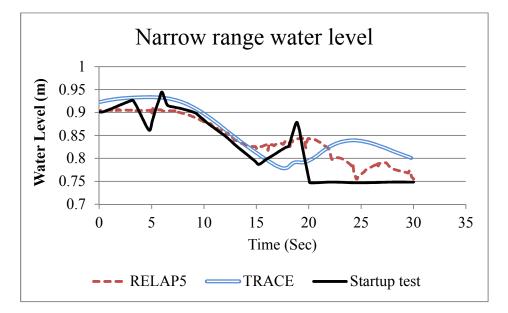


Figure 33 Narrow Range Water Level Variation during the 94% FWPT Transient

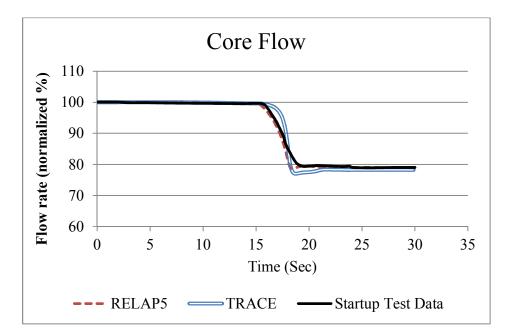


Figure 34 Core Flow Variation during the 94% FWPT Transient

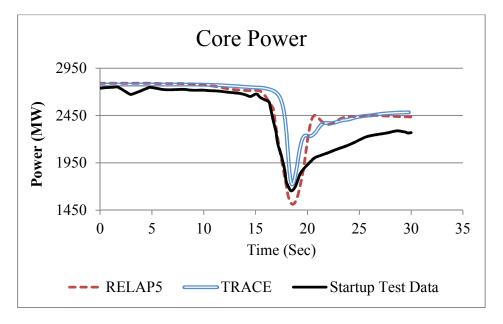


Figure 35 Core Power Variation during the 94% FWPT Transient

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.

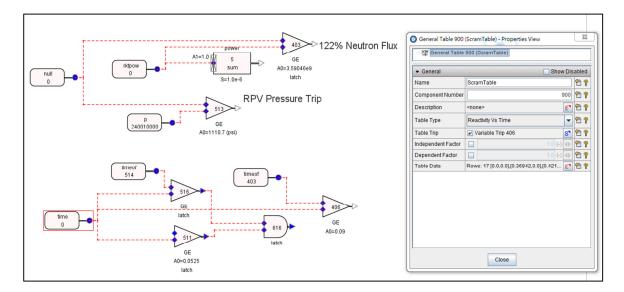


Figure 36 Controlling System of Reactor Scram in MSIVC Hypothetical Accident

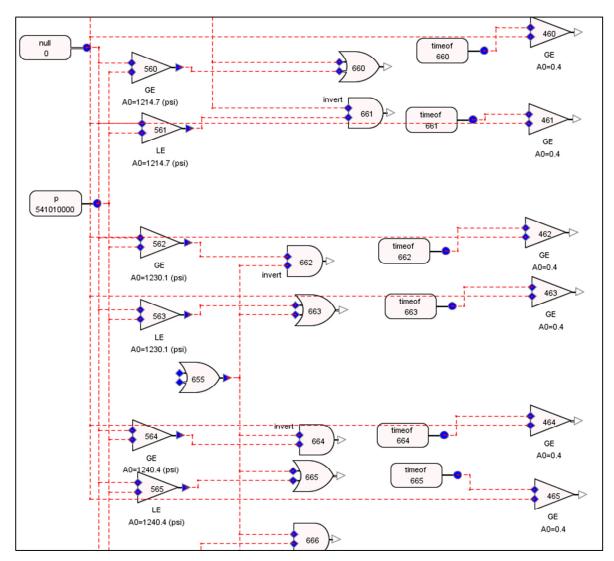


Figure 37 Controlling System of Safety Valves in MSIVC Hypothetical Accident

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	

Table 8	Events Sequence and Com	parison of MSIVC Hypothetical Accident
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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 variation did not triggered the setpoints 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.

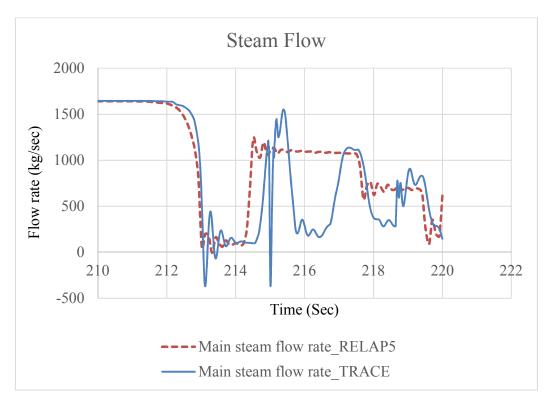


Figure 38 Steam Flow Variation during the MSIVC Hypothetical Accident

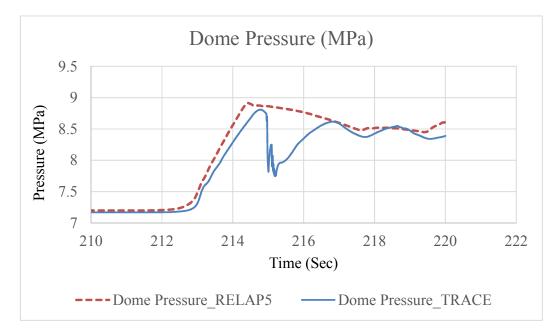


Figure 39 Dome Pressure Variation during the MSIVC Hypothetical Accident

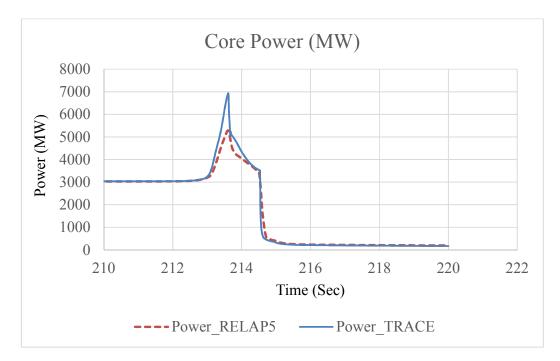


Figure 40 Core Power Variation during the MSIVC Hypothetical Accident

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

Hypothetical Event Sequence

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.

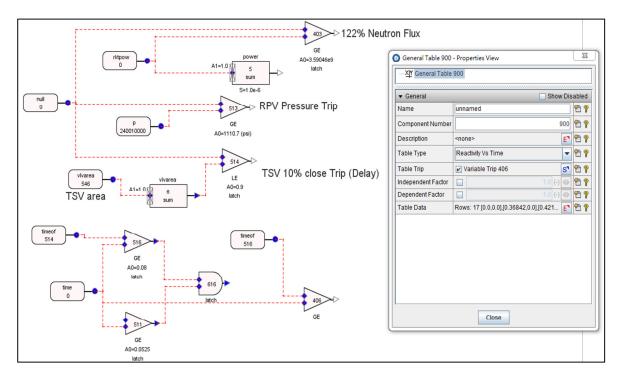


Figure 41 Controlling System of Reactor Scram in TTBF Hypothetical Accident

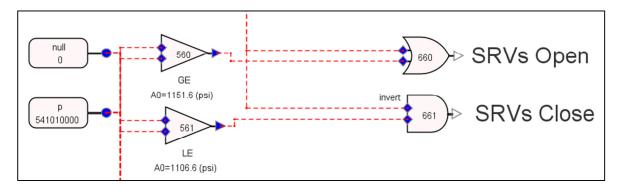


Figure 42 Controlling System of Relief Valves in TTBF Hypothetical Accident

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	

Table 9 Events Sequence and Comparison of TTNB Hypothetical Accident

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.

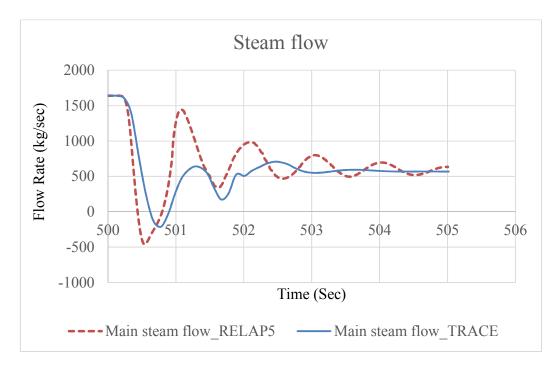


Figure 43 Steam Flow Variation during the TTBF Hypothetical Accident

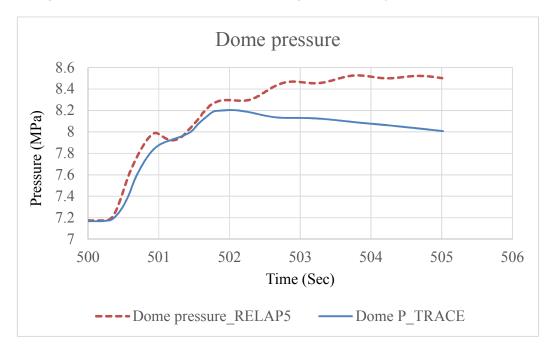


Figure 44 Dome Pressure Variation during the TTBF Hypothetical Accident

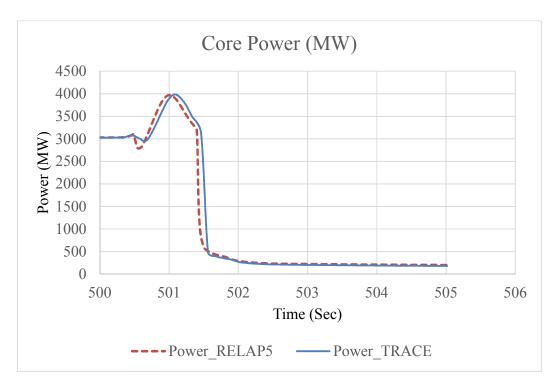


Figure 45 Core Power Variation during the TTBF Hypothetical Accident

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.

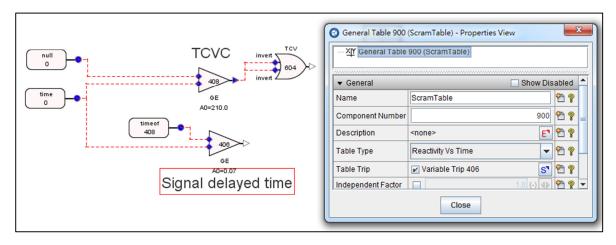


Figure 46 Controlling System of Reactor Scram in LRNB Hypothetical Accident

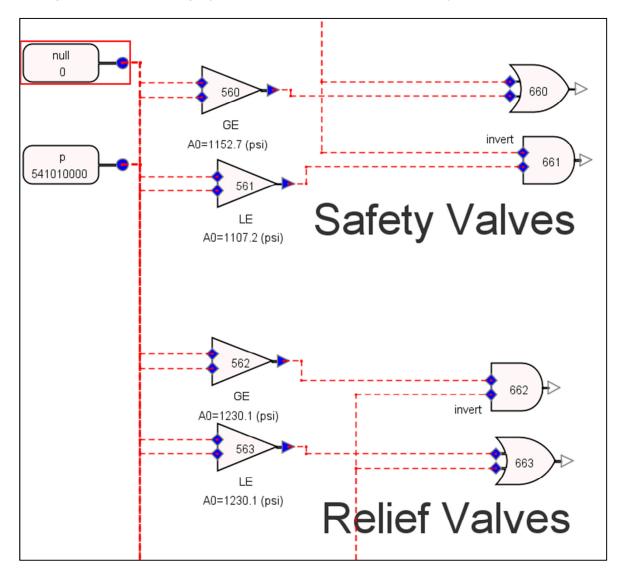


Figure 47 Controlling System of Safety/Relief Valves in LRNB Hypothetical Accident

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	

Table 10	Events Sequence and	Comparison of LRNB Hypothetical Accident
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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.

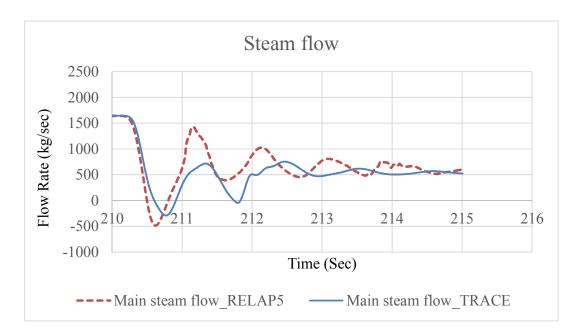


Figure 48 Steam Flow Variation during the LRNB Hypothetical Accident

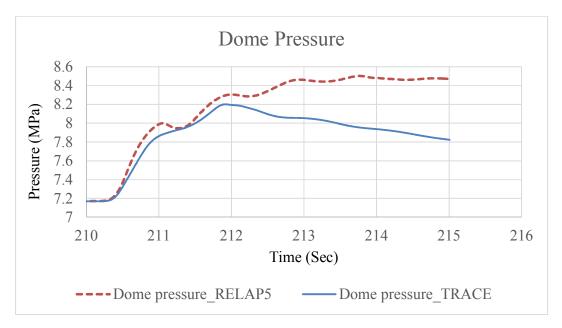


Figure 49 Dome Pressure Variation during the LRNB Hypothetical Accident

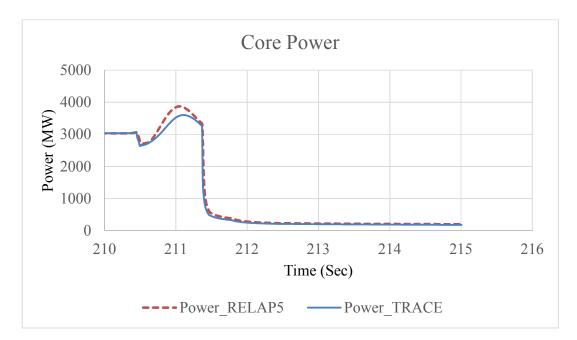


Figure 50 Core Power Variation during the LRNB Hypothetical Accident

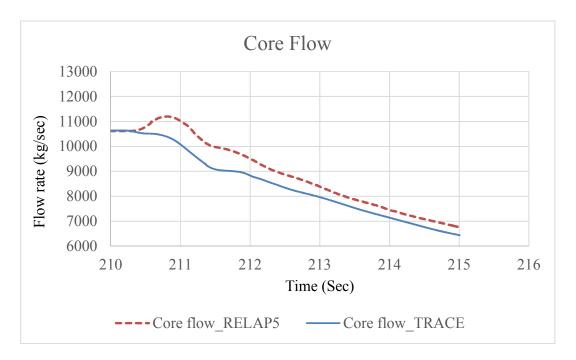


Figure 51 Core Flow Variation during the LRNB Hypothetical Accident

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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NRC FORM 335 (12-2010) NRCMD 3.7 BIBLIOGRAPHIC DATA SHEET (See instructions on the reverse)	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/IA-0464				
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Power Plant with SNAP Interface	March	2016			
	4. FIN OR GRANT NUMBER				
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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.)		LITY STATEMENT			
Measurement Uncertainty Recapture (MUR) Stretch Power Uprate (SPU)		UNIIMITED			
RELAP5/MOD3.3	(This Page)				
Main Steam Isolation Valve (MSIV)		nclassified			
Kuosheng Nuclear Power Plant (NPP)	(This Report				
		nclassified R OF PAGES			
	16. PRICE				

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