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Thermal Hydraulic and Fuel Rod Mechanical Combination Analysis of Kuosheng Nuclear Power Plant with RELAP5 MOD3.3/FRAPTRAN/Python in SNAP Interface

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ABSTRACT

After the measurement uncertainty recapture power uprates, Kuosheng nuclear power plant (NPP) was uprated the power from 2894 MWt to 2943 MWt. For this 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 analyses should be done for ensuring that the power plant could maintain stability in higher power operating conditions. In this research, three overpressurization transients scenario including main steam isolation valves closure, turbine trip with bypass failure and load rejection with bypass failure would be performed by RELAP5 MOD3.3 code. Further, the thermal hydraulic properties of the reactor core will be transferred as the boundary conditions of FRAPTRAN code. With the boundary conditions from RELAP5 code, the fuel rod mechanical properties during the transient could be determined. In this research, the SNAP interface is applied so that the transferring process between RELAP5 and FRAPTRAN code can be completed automatically with Python Job Stream. That is, the researchers need not calculate and transfer the thermal hydraulic boundary conditions for FRAPTRAN analysis manually.

FOREWORD

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

Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program). 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, a RELAP5/MOD 3.3 and FRAPTRAN 1.5 combination model of Kuosheng NPP which can transfer the data results of RELAP5 code into FRAPTRAN input deck automatically was developed with SNAP interface.

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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. 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 analyses should be done for ensuring that the power plant could maintain stability in higher power operating conditions. In this research, three overpressurization transients scenario including main steam isolation valves closure, turbine trip with bypass failure and load rejection with bypass failure would be performed by RELAP5 MOD3.3 code. Further, the thermal hydraulic properties of the reactor core will be transferred as the boundary conditions of FRAPTRAN code. With the boundary conditions from RELAP5 code, the fuel rod mechanical properties during the transient could be determined. In this research, the SNAP interface is applied so that the transferring process between RELAP5 and FRAPTRAN code can be completed automatically with Python Job Stream. That is, the researchers need not calculate and transfer the thermal hydraulic boundary conditions for FRAPTRAN analysis manually.

ABBREVIATIONS

BWR	Boiling-Water Reactor
BPV	Bypass valve
INER	Institute of Nuclear Energy Research Atomic Energy Council, R.O.C.
kg	kilogram(s)
kW	kilowatt(s)
LRBF	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
NRWL	Narrow Range Water Level
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
TTBF	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]. Few years later, 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 ratio, 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, 4, 5]. 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 [6]. 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.

Fuel Rod Analysis Program Transient (FRAPTRAN) is a light water reactor fuel rods simulation code which is conducted by U.S. NRC and developed by Pacific Northwest National Laboratory (PNNL) since 1997. FRAPTRAN is developed to calculate the single fuel rod behavior during transients and hypothetical accidents at burnup level up to 62 gigawatt-days per metric ton of Uranium (GWd/MTU) [7]. In this research, FRAPTRAN version 1.5 was applied. With FRAPTRAN 1.5, the important criteria such as cladding temperature, cladding hoop strain, fuel enthalpy and radial gap can be determined. However, some long term properties such as fuel densification, swelling, radiation growth and cladding creep are not concluded in FRAPTRAN code. Hence, the FRAPCON-3 code is also applied in this research.

FRAPCON-3 is also conducted by U.S. NRC and developed by PNNL. Different from FRAPTRAN code, FRAPCON-3 analyzes the fuel rod behavior when the power and coolant conditions change slowly. The calculation results supply several fuel rods properties and burnup information after a long-term operation [8]. The restart file from FRAPCON-3 analysis can be the initial conditions of FRAPTRAN analysis. With this restart file, some long-term operating properties can be concerned in FRAPTRAN transient analysis. In this research, FRAPTRAN code referenced the restart file of 18-month fuel operation which was calculated by FRAPCON-3 code.

Symbolic Nuclear Analysis Package (SNAP) is an interface of NPP analysis codes which developed by US NRC and Applied Programming Technology, Inc. Different from the traditional input deck in ASCII files, the graphical control blocks and thermal hydraulic connections make researches comprehend the whole power plant and control system more easily [9]. Due to these advantages, the RELAP5/MOD3.3 model of Kuosheng NPP was developed with SNAP interface.

In the past, the TRACE thermal hydraulic and FRAPTRAN fuel rod mechanical property combined analysis had been performed [10, 11]. In this kind combination analysis, the data results cannot only show the thermal hydraulic properties but also the fuel rod deformation during the transient. However, the data transferring process from TRACE to FRAPTRAN was complicated and time-consuming. In this research, a developed RELAP5/MOD 3.3 model [12, 13] was applied for the thermal hydraulic data. However, the automatic transferring process for FRAPTRAN input deck was developed. Thermal hydraulic data from RELAP5/MOD 3.3 analysis can be easily converted into FRAPTRAN input deck and the whole both of the thermal hydraulic and fuel rod mechanism analysis can be finished with one model in SNAP interface.

2 MODEL ESTABLISHMENT

The performance of the RELAP5 model had been verified and assessed last year [13]. However, the thermal hydraulic analysis of Kuosheng NPP performed last year could only obtain the reactor vessel and pipeline information during the transient. Information of fuel rod mechanical variations is unobtainable for the developed RELAP5 model. As a result, combined analysis of RELAP5 and FRAPTRAN is required for more detailed information of fuel rods during transient. In this research, analysis models of different codes including RELAP5/MOD 3.3, FRAPTRAN 1.4 and FRAPCON 3.4 are developed in SNAP interface which can directly and conveniently control the job streams of these codes. At first, the RELAP5 model of Kuosheng NPP was developed in SNAP interface singly. Then, this verified model would be referenced by "Eng-template" model which can combine different code analysis in SNAP interface. With this useful model type of SNAP interface, the combination of these two codes becomes possible. From the experience of our group, the combination of thermal hydraulic and fuel rod performance code was successfully performed with TRACE and FRAPTRAN codes. The same analysis flow is followed in this research. However, different form the TRACE-FRAPTRAN analysis, the RELAP5-FRAPTRAN analysis can be applied with single job submitting. That is, the thermal hydraulic information of RELAP5 code can be transferred into FRAPTRAN input deck automatically. To complete such job stream, the python script which can extract and transfer thermal hydraulic information is developed in the SNAP interface. The flow chart of this combination model is shown in Figure1.

2.1 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 2. 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. 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.

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. 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.

To simplify the control model and reduce the computer time, the scram control systems were developed according to each case respectively. However, the scram reactivity feedback 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.

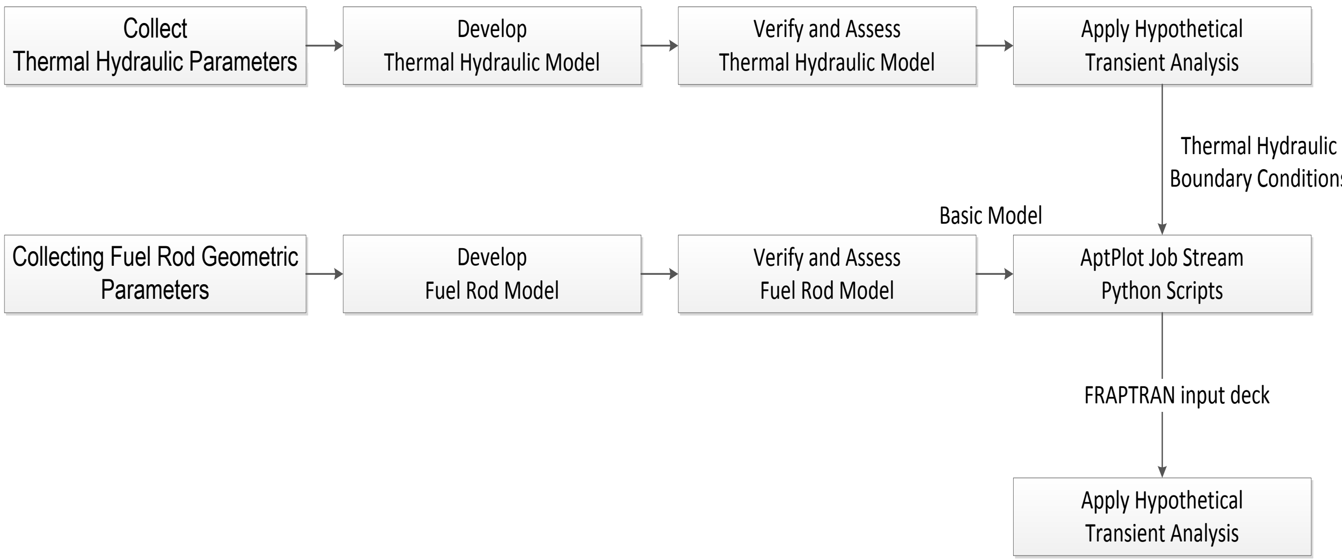

Figure 1 Flow chart of thermal hydraulic and fuel rod mechanism analysis model

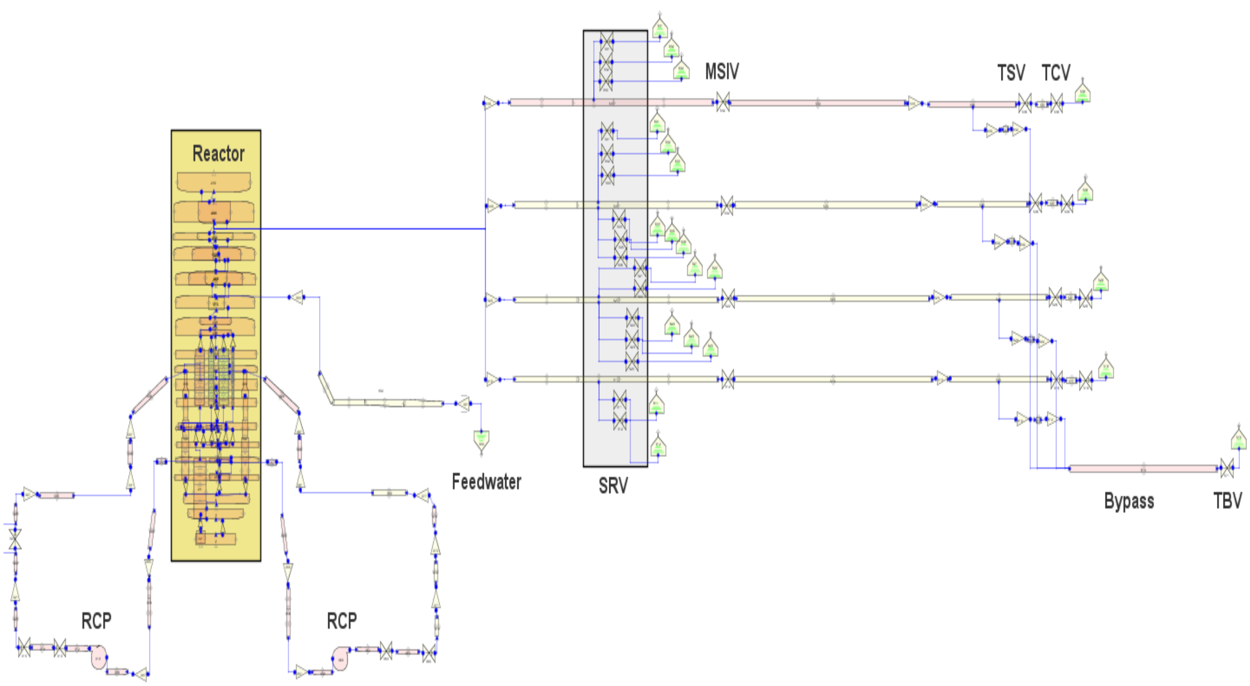

Figure 2 Overview of the Kuosheng NPP RELAP5/MOD3.3 model

2.2 Overview of the FRAPTRAN analysis model

The fuel rod geometric information was shown in Table 1. Some steady state fuel rod information such as burn-up value and fission gas is calculated by FRAPCON-3. The long-term information of FRAPCON-3 data results will be combined into the restart file, which can be read and calculated by FRAPTRAN code. As mentioned before, the boundary conditions required would be performed by TRACE and transferred to FRAPTRAN code manually.

In the FRAPTRAN analysis, the time interval was 0.01 second. The analysis time was set up according to the hypothetical accidents respectively. The gap pressure and temperature convergence are both 0.001. The maximum temperature variation between two iterations is 6K. After setting up the convergence index, the limitation of iteration numbers is also required to prevent the computer crashing if the calculations could not meet the convergence index. In this research, both the limitations of iterations in transient and steady state are set as 100.

In this research, the fuel rod nodes were divided into radial and axial directions. For the radial directions, it can be further divided into two settings including nodes inside the fuel pellet and nodes inside the cladding. As shown in Figure 3 and Figure 4, there are 13 radial nodes with equal interval started from the center of the fuel pellet and ended at the surface of the fuel pellet. Besides, same in the radial directions, there are two nodes inside and outside the cladding. However, it should be noticed that the radials nodes would be only calculated in the code iterations. The output data would be normalized the radial nodes information. For the axial direction, in this research, the fuel rod is divided into 12 nodes with equal interval. The output information of temperature, pressure, strain, stress and so on will be recorded and plotted mainly according to these 12 axial nodes.

Table 1 Fuel rod geometry

Parameters	Input codes	value	Unit	Notes
Length	RodLength	12.45	Feet	
Diameter	RodDiameter	0.03608	Feet	
Gap thickness	gapthk	0.00031	Feet	Replaced with FRAPCON data
Upper spring coils	ncs	59	N/A	
Upper spring height	spl	0.7513	Feet	
Upper spring diameter	scd	0.02583	Feet	
Upper spring wire diameter	swd	0.00425	Feet	
Cold works	coldw	0.5	N/A	
Cladding roughness	roughc	1	um	
Helium fraction	gfrac(1)	1	N/A	
Fill pressure	gappr0	74.7	psia	
Pellet diameter	FuelPelDiam	0.0305	Feet	Replaced with FRAPCON data
Pellet height	pelh	0.0366	Feet	Replaced with FRAPCON data
Fuel surface roughness	roughf	1	um	Replaced with FRAPCON data
Pellet density	frden	0.945	N/A	
Fuel sintering temperature	tsntrk	1773	K	
Fuel grain size	fgrns	10	um	

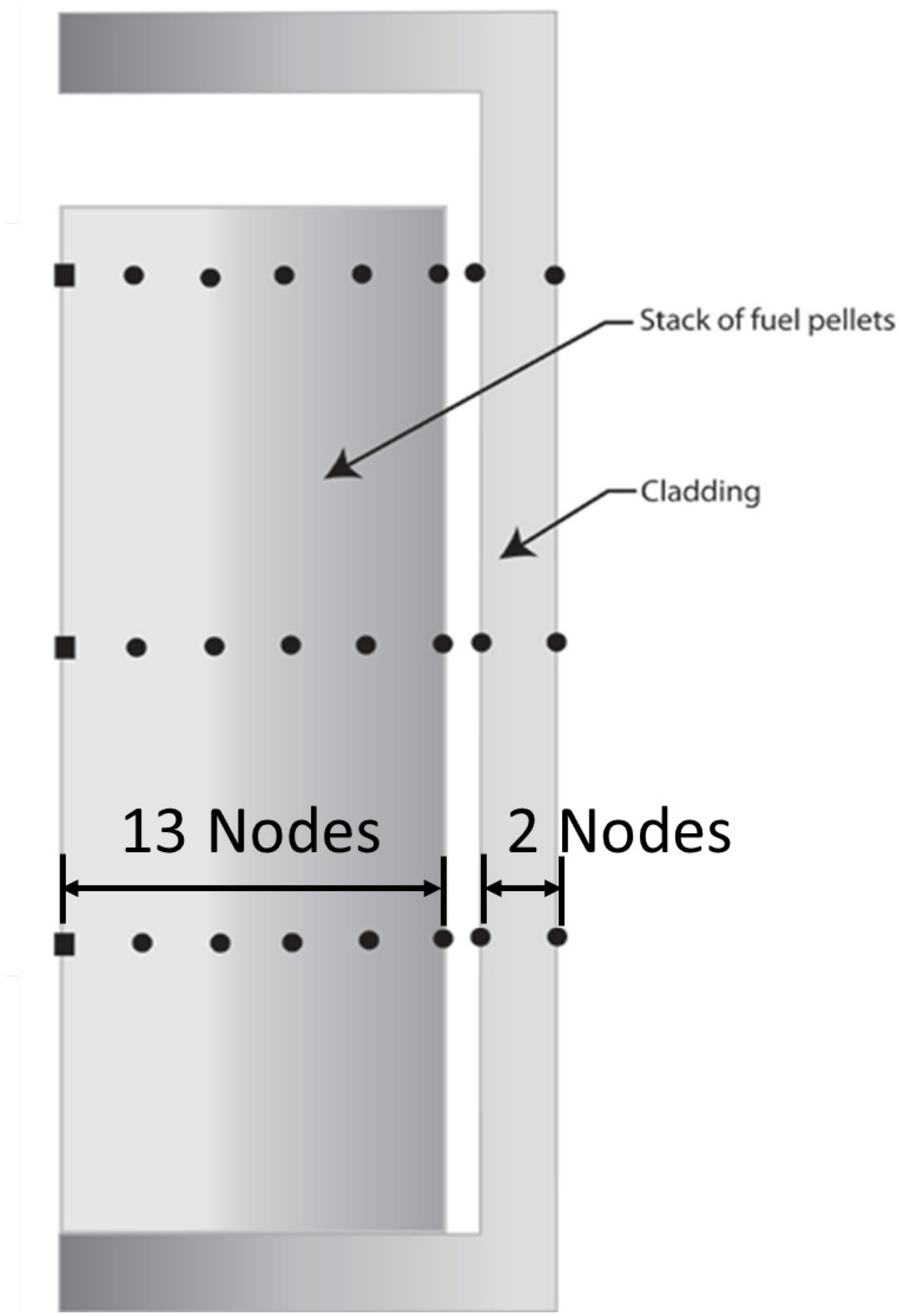

Figure 3 Setting of radial nodes in FRAPTRAN model

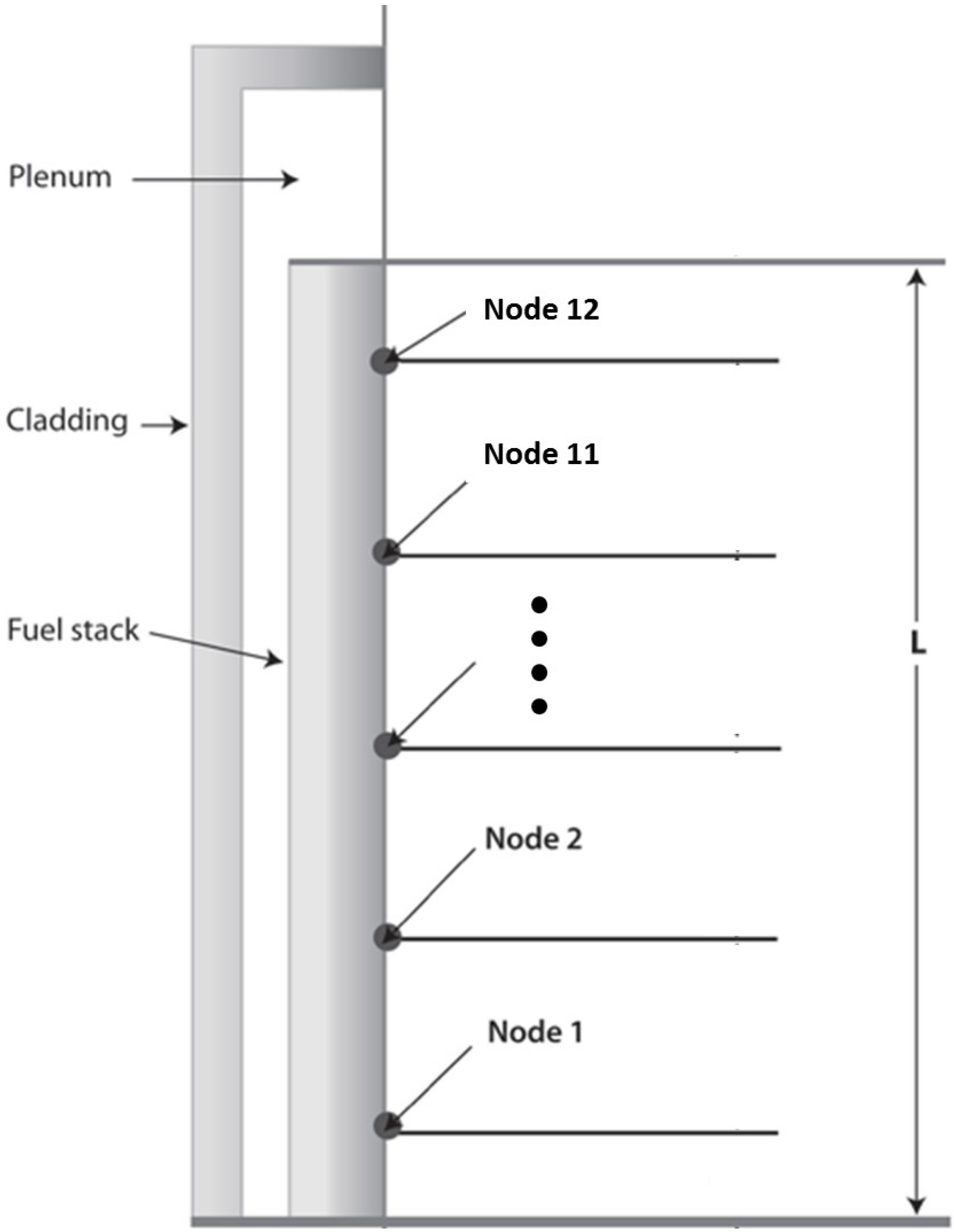

Figure 4 Setting of axial nodes in FRAPTRAN model

2.3 Job stream assignment in SNAP interface

In the SNAP, there is a model named "Eng-template" which allows users create analysis process with different kind of computer code. As developing this model, the reference model with single computer code should be developed previously. As shown in Figure 5, in the Eng-template model, there are two different computer analysis model created including the RELAP5 and FRAPCON reference model. To be noticed, the developed FRAPTRAN model is not referenced in this way because there are a lot of boundary conditions which are from the RELAP5 results need to be modified with python scripts. After modified by the python scripts, ASCII input deck will be transferred into FRAPTRAN job stream to do the next analysis about fuel rods.

To retrieve thermal hydraulic data results from the RELAP5 analysis efficiently, the AptPlot job stream is also developed in this model. In this job stream, there are four thermal hydraulic data retrieved including the power history, dome pressure, heat transfer coefficient and core temperature. In the RELAP5 model, there are control blocks which extract power history in unit MW and dome pressure in unit psi respectively; hence, the data channels "cntrlvar-5" (for power history) and "cntrlvar-650" (for dome pressure) are described in AptPlot job stream settings as shown in Figure 6. Moreover, the fuel rods are represented by the heat structure in RELAP5 code. As a result, the data channels "hthtc-" and "tempf-" are described in AptPlot job stream node by node to retrieve the heat transfer coefficient and coolant temperature for FRAPTRAN analyses, as shown in Figure 7. To be noticed, there are three group of fuel assemblies developed in the RELAP5 model. For safety, the hottest fuel assembly heat structure (1621) information is applied. To transfer these data results retrieved by the AptPlot job stream, the output file type is defined as "ASCI" file, which will make the python job stream conversion more efficiently.

In the python job stream, the input file type can be defined manually. In this case, there are 33 input files including 1 power history, 1 dome pressure, 1 FRAPTRAN input sample file, 15 heat transfer coefficient and 15 temperature data results. All these input files are respectively connected to the corresponding extraction data files from the AptPlot job stream as shown in Figure 8. Moreover, users should also determine what kind of output file type is desired. In this case, the FRAPTRAN input file with filename extension ".inp" is chosen. After setting up the input and output characters of the python job stream, the python code can be developed in the "python scripts" dialog (shown in Figure 9). At the beginning of the python scripts, the data files from the AptPlot job stream should be read into the memory so that the data results can be processed in the following steps. To read the data file from the previous job stream, there is a python function "get_input_file()" which allows users transfer a data file into a python variable. Then, the variable can be opened through the python "*open()*" function. In addition to the thermal hydraulic data results files, the FRAPTRAN sample input deck should also be included in the job stream. This file will be introduced into SNAP with external file which can also be input by the python job stream. With the sample input deck file, data results from the RELAP5 code can be written into the corresponding input structure of the FRAPTRAN input deck. Moreover, it should be noted that the units of data results from REALP5 code may different from the FRAPTRAN input deck. For instance, the unit of power from the RELAP5 code is MW for the whole reactor core while the unit of power in the FRAPTRAN deck is kw/ft for single fuel rod. A proper unit conversion should be concerned in corresponding cases.

After combined with the thermal hydraulic data results and fuel rod design input deck, the output file of python job stream will be delivered to FRAPTRAN code to do the further analysis. Hence, the output file type setting of python job stream should be defined as FRAPTRAN input which can be connected to FRAPTRAN job stream. In addition, the burn-up value is also an important concerned in the FRAPTRAN analysis. In this Eng template, the "restart" file was generated with

FRAPCON 3.4 code and connected to the FRAPTRAN job stream. In this research, 18-month burn-up calculation is assumed in FRAPCON, which is the same period of re-fueling.

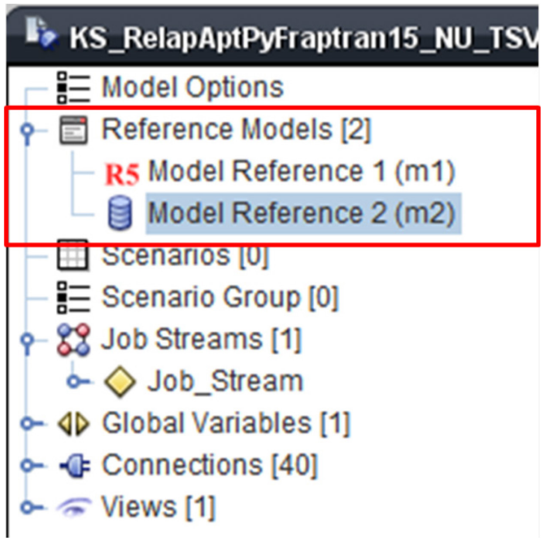

Figure 5 Reference model of the combination analysis

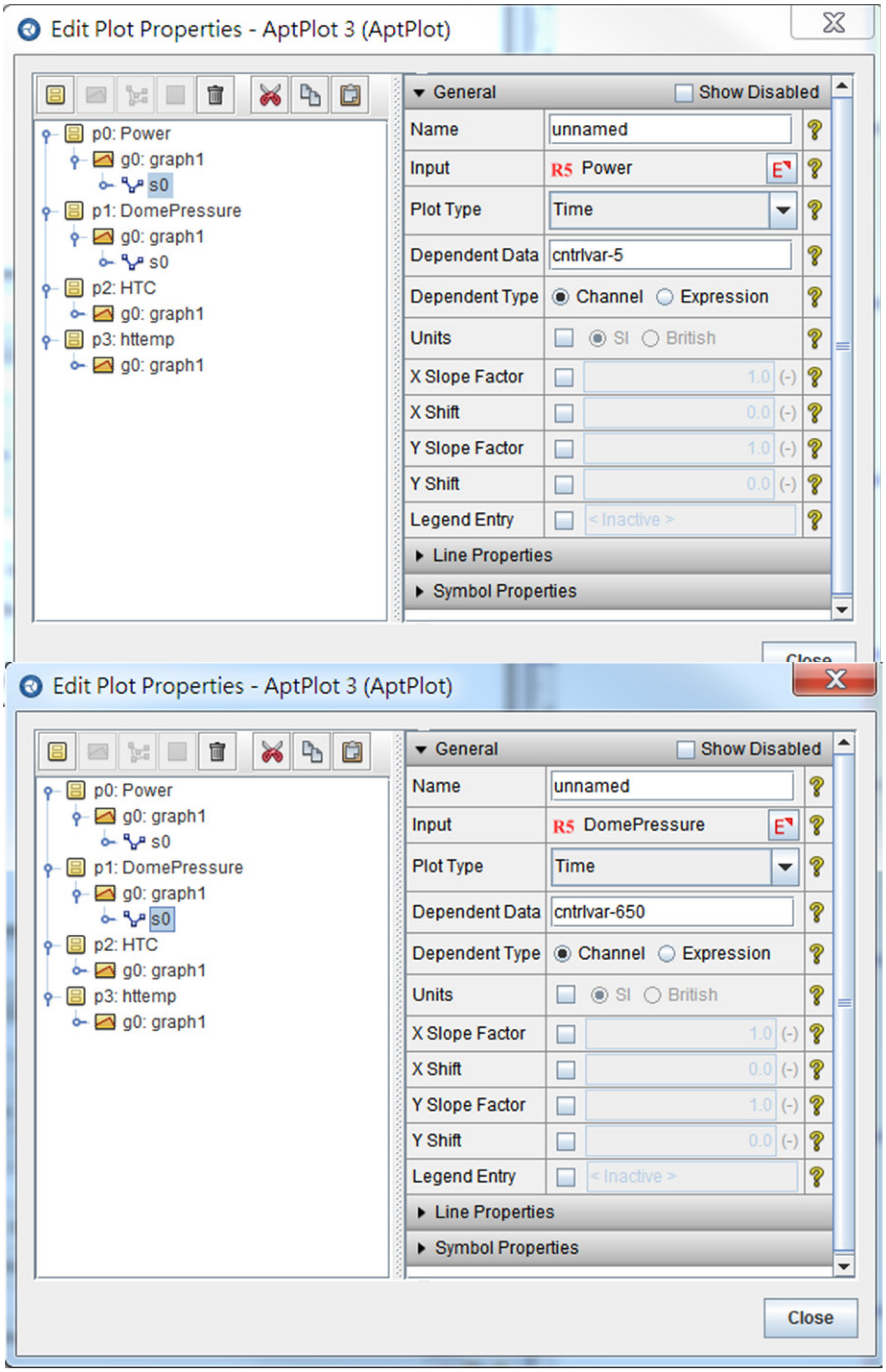

Figure 6 AptPlot job stream setting for power history and dome pressure

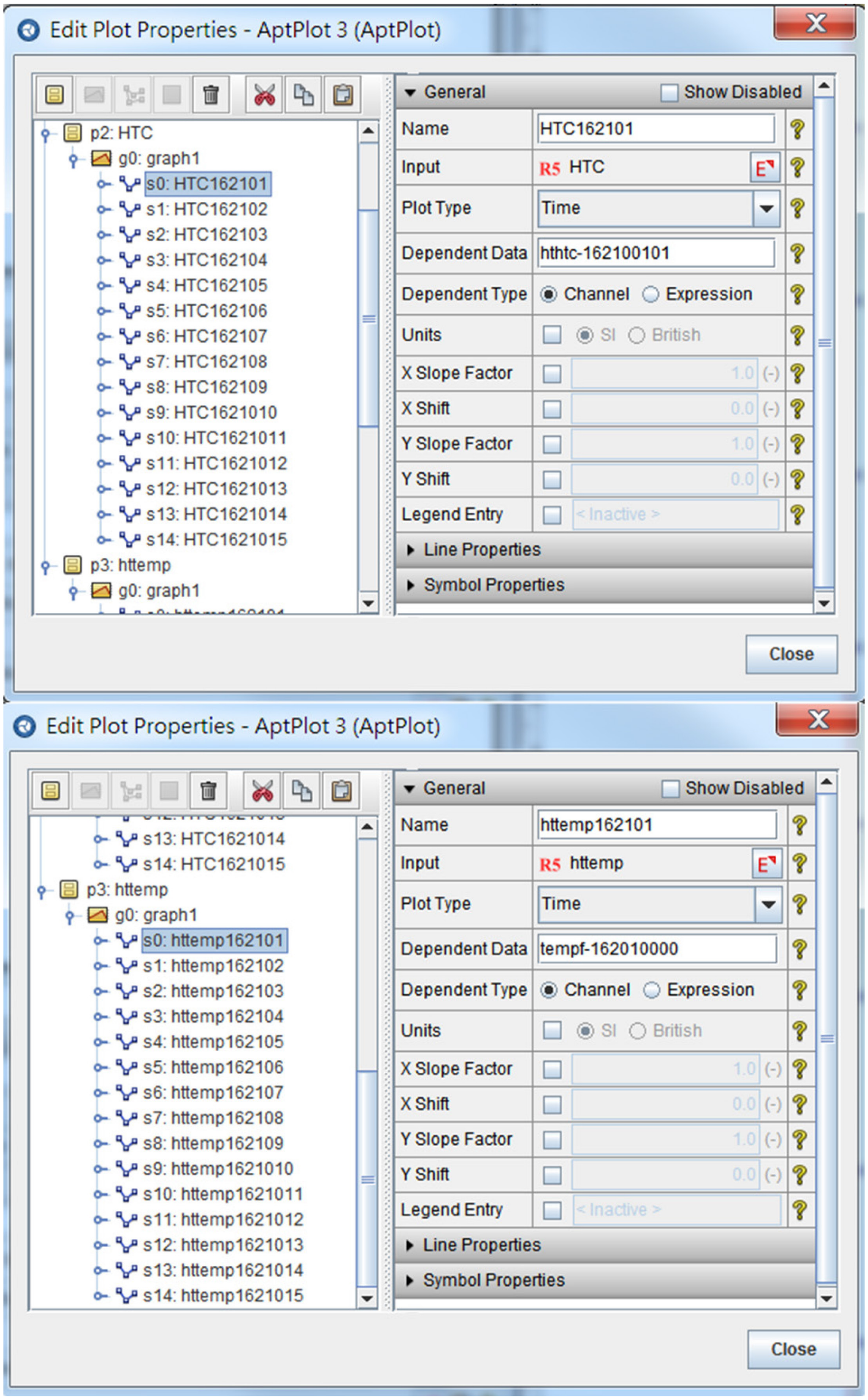

Figure 7 AptPlot job stream setting for heat transfer coefficient and coolant temperature

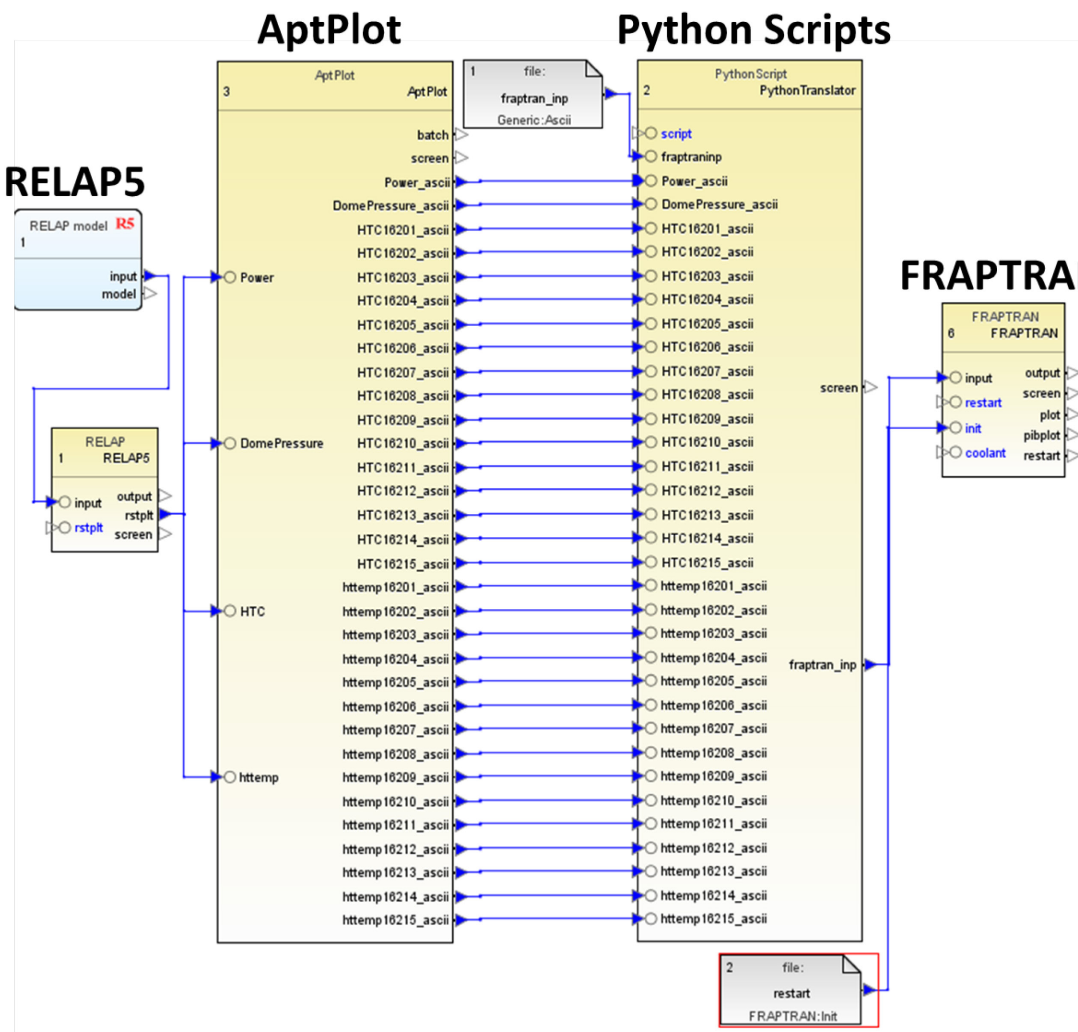

Figure 8 Job stream connection

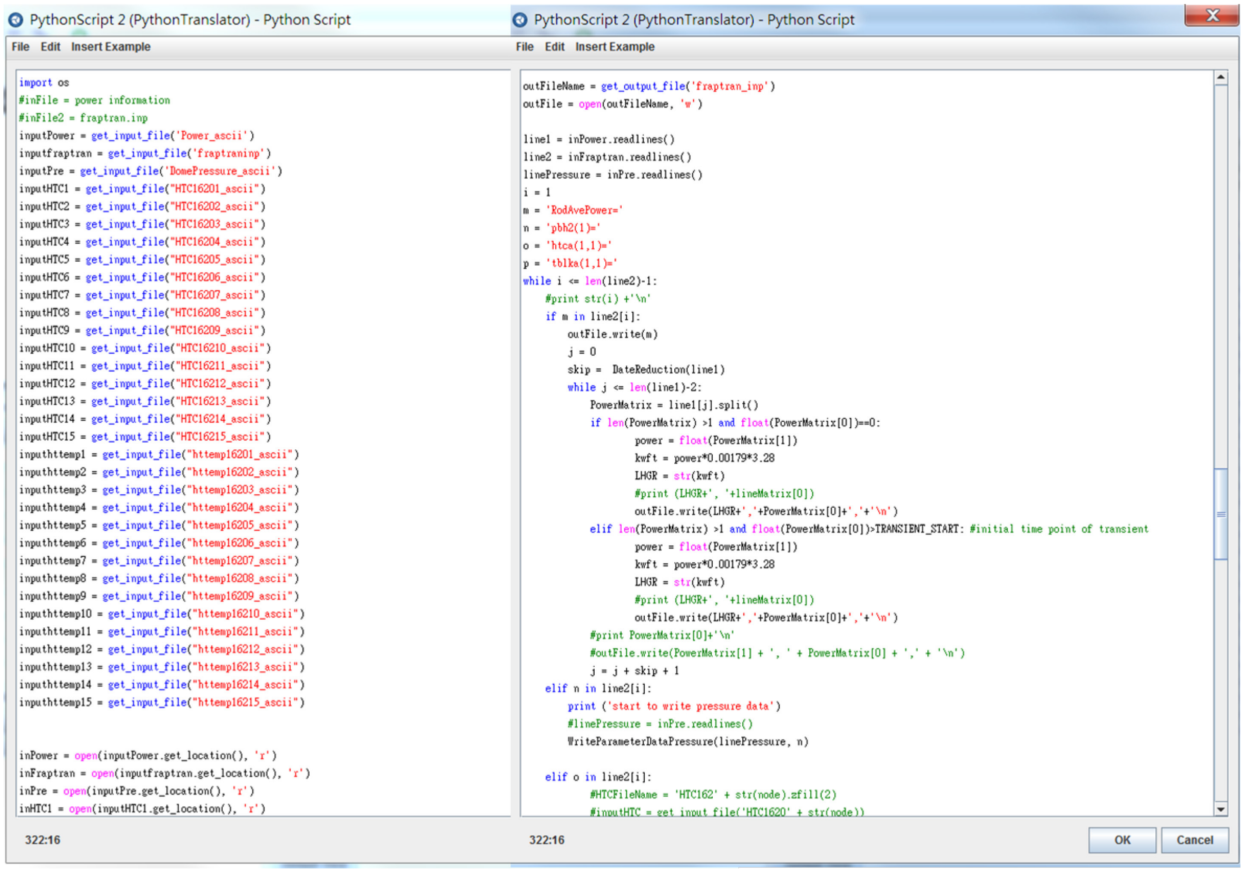

Figure 9 Python scripts dialog

3 HYPOTHETICAL ACCIDENT

To verify the applicability of RELAP5-FRAPTRAN analysis model, hypothetical accidents of Kuosheng NPP were performed. With these accidents performed in the new model, there are two main advantages. First, these accidents had been performed with TRACE and FRAPTRAN codes (with manually data transferring) so there were a lot of data results which could be compared. Second, the transient duration of these hypothetical accidents are short, which are convenient for model modification.

As mentioned above, a series of overpressurization transient test of TRACE code had been performed before by our group. In this research, same hypothetical accidents will be performed again by RELAP5 code and then these data results will be transferred as FRAPTRAN input deck with python job stream in SNAP interface. Hence, the initial conditions such as power ratio, feedwater flow rate, steam flow rate and dome pressure of RELAP5 code should be same as that of TRACE code so that the comparison would be useful and meaningful. The hypothetical accidents assumption and event sequence are described below.

Table 2 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	1641.2	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
NRWL	(m)	0.934	0.934	0.933

3.1 <u>Main Steam Line Isolation Valves Closure with Bypass Failure</u>

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 10. 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 11, 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 3 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.

Table 3 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 ratio 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	

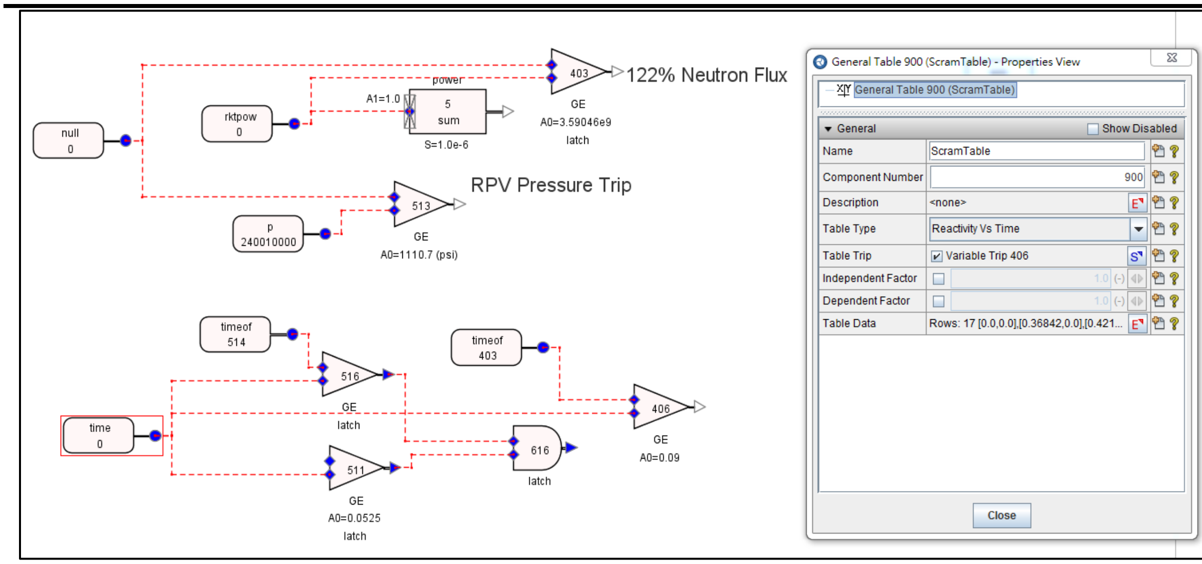

Figure 10 Controlling system of reactor scram in MSIVC hypothetical accident

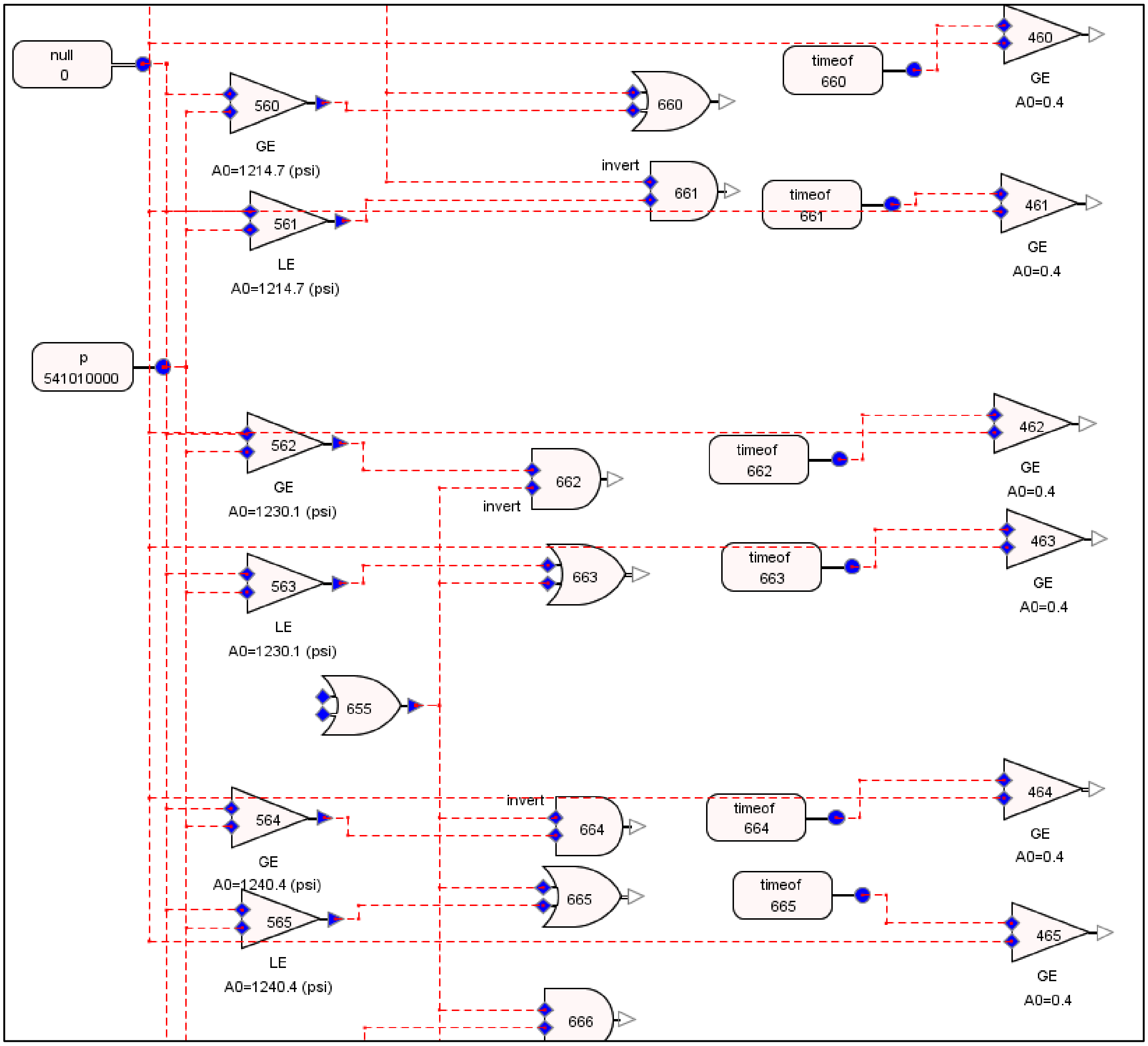

Figure 11 Controlling system of safety valves in MSIVC hypothetical accident

3.2 Turbine Trip with Bypass Failure

In the hypothetical accident of TTBF, 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 TTBF 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 12.

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 ratio 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 13. Further, the safety relief valves would fully open in 0.15 second. Once the safety relief valves opened, the dome pressure and the power plants would be under controlled. Table 4 listed the event sequences and important setpoints of the TRACE model during TTBF transient.

Table 4 Events sequence and comparison of 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	

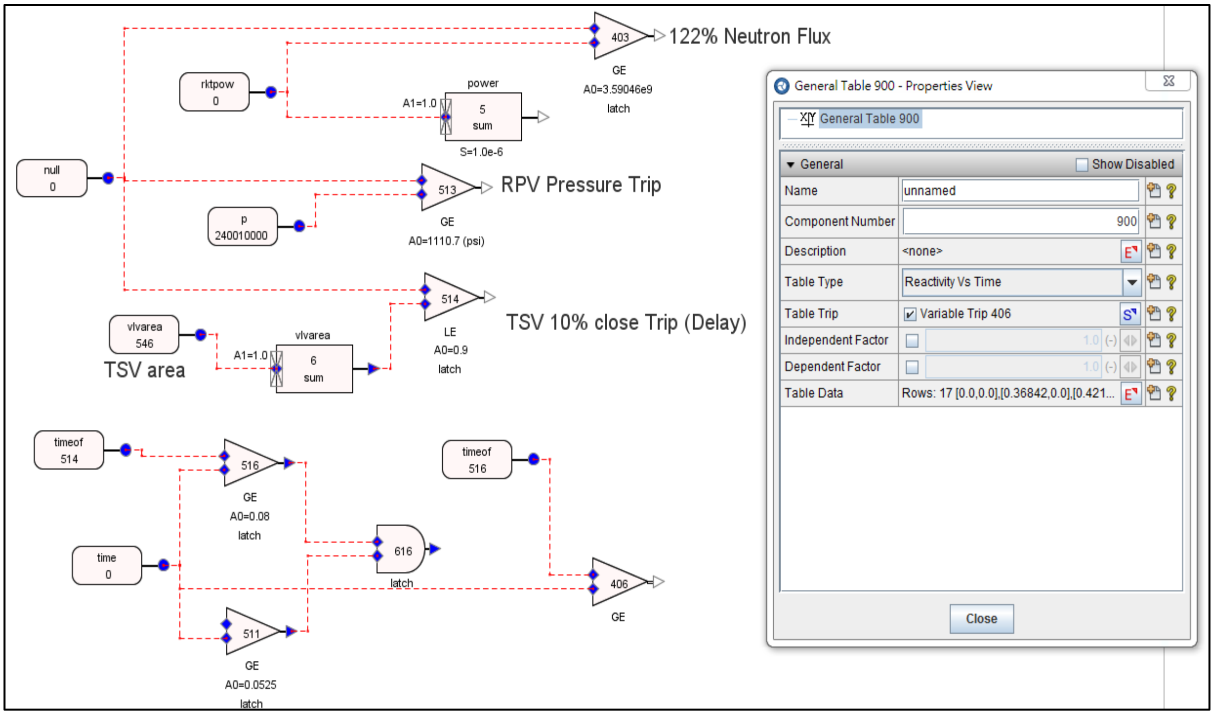

Figure 12 Controlling system of reactor scram in TTBF hypothetical accident

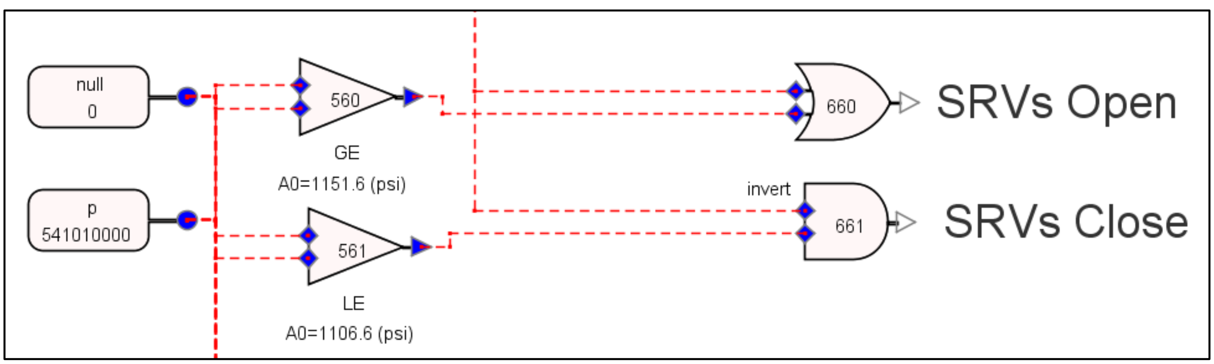

Figure 13 Controlling system of relief valves in TTBF hypothetical accident

3.3 Load Rejection with Bypass Failure

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 14. 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 15. Same as previous two cases, once the safety/relief valves open, the dome pressure decreased and the NPP was under controlled. Table 5 lists the important setpoints and transient events of the TCVC hypothetical accident.

Table 5 Events sequence and comparison of LRBF 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	

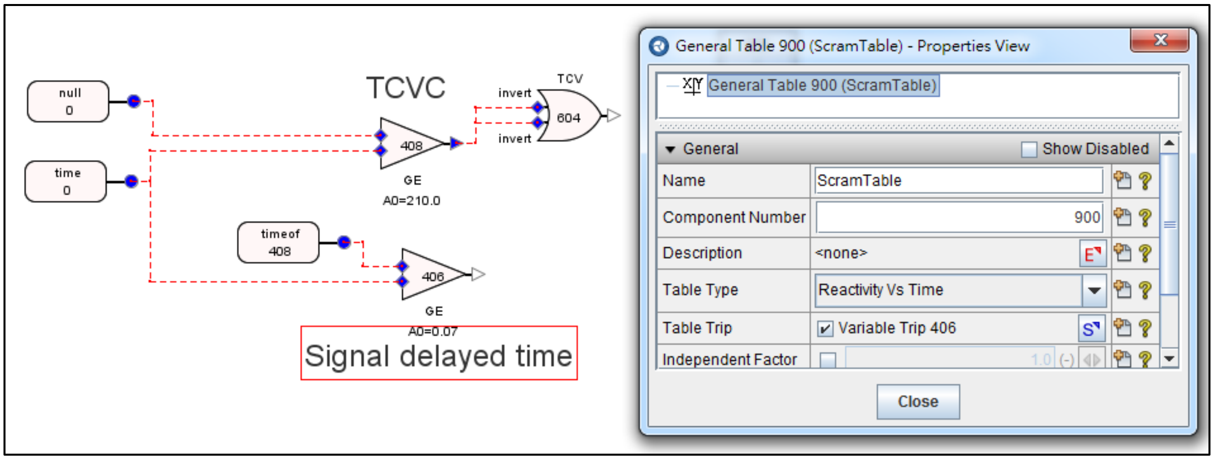

Figure 14 Controlling system of reactor scram in LRBF hypothetical accident

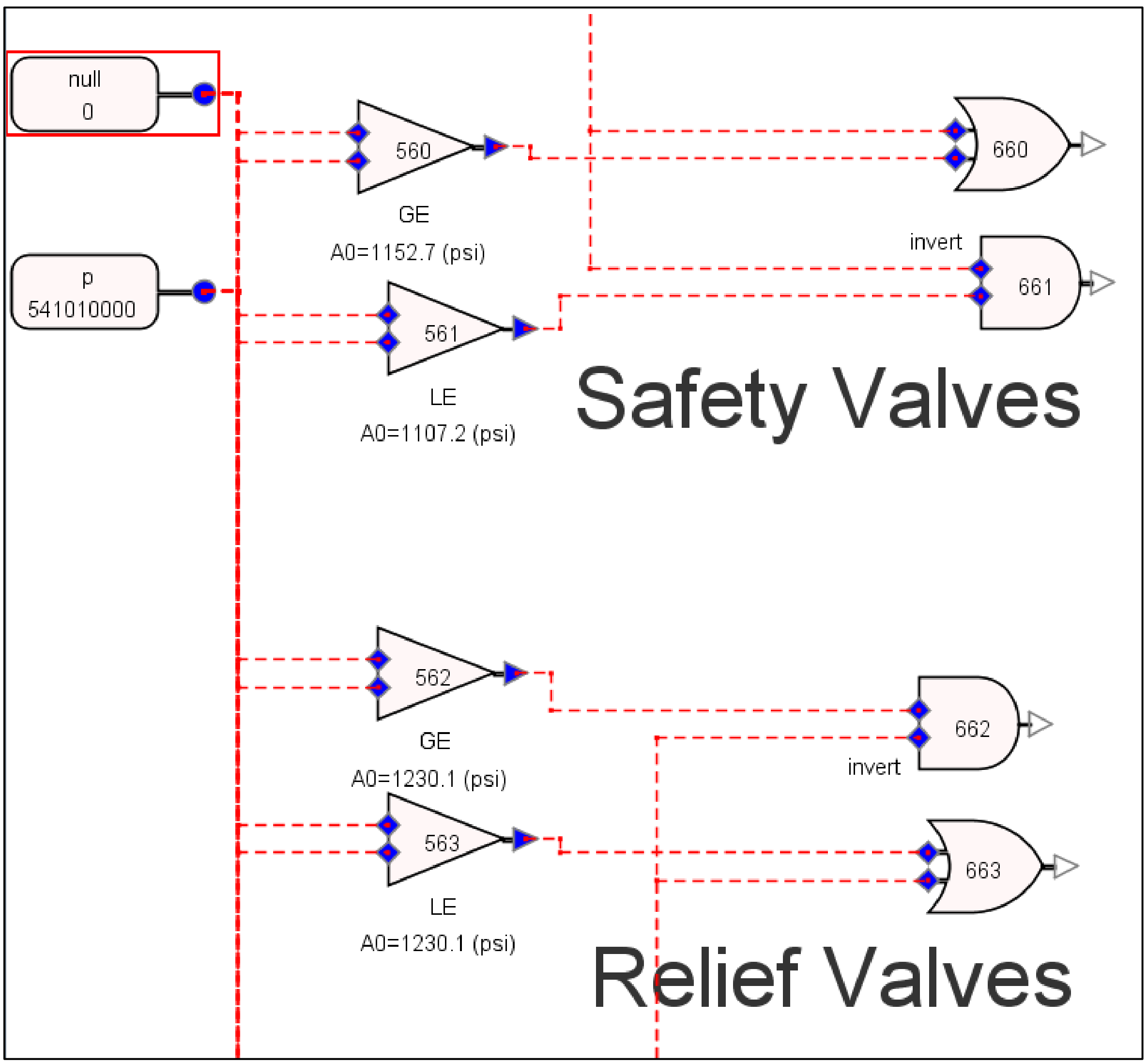

Figure 15 Controlling system of safety/relief valves in LRBF hypothetical accident

4 RESULTS

In each hypothetical accident, there are two parts of data results will be illustrated. For the previous part, the analysis results of TRACE and RELAP5 will be compared to realize the computational difference of two codes. In fact, the comparison had been described in the NUREG report published by our group last year. However, to explain the FRAPTRAN analysis results comparisons more clearly, analysis results comparisons of TRACE and RELAP5 codes are described again in current report.

The main FRAPTRAN analysis results of current report will be described in the second part. All the 12 nodes data results of regulatory criteria including cladding temperature, cladding hoop strain and fuel enthalpy will be plotted. With these figures, it can be easily determined that if the fuel rod keeps good integrity during the transient. In addition, the FRAPTRAN analysis results which boundary conditions were from TRACE analysis results will also be compared to the FRAPTRAN analysis results of current research.

4.1 Main Steam Line Isolation Valves Closure with Bypass Failure

4.1.1 Thermal Hydraulic Analysis data results

Figure 16 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 17. Hence, the void fraction inside the reactor core decreased which increased the power greatly as shown in Figure 18. 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 trigger the setpoints of safety valves closure in the RELAP5 model. Hence, as shown in Figure 17, the reactor vessel in the RELAP5 model maintained 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.

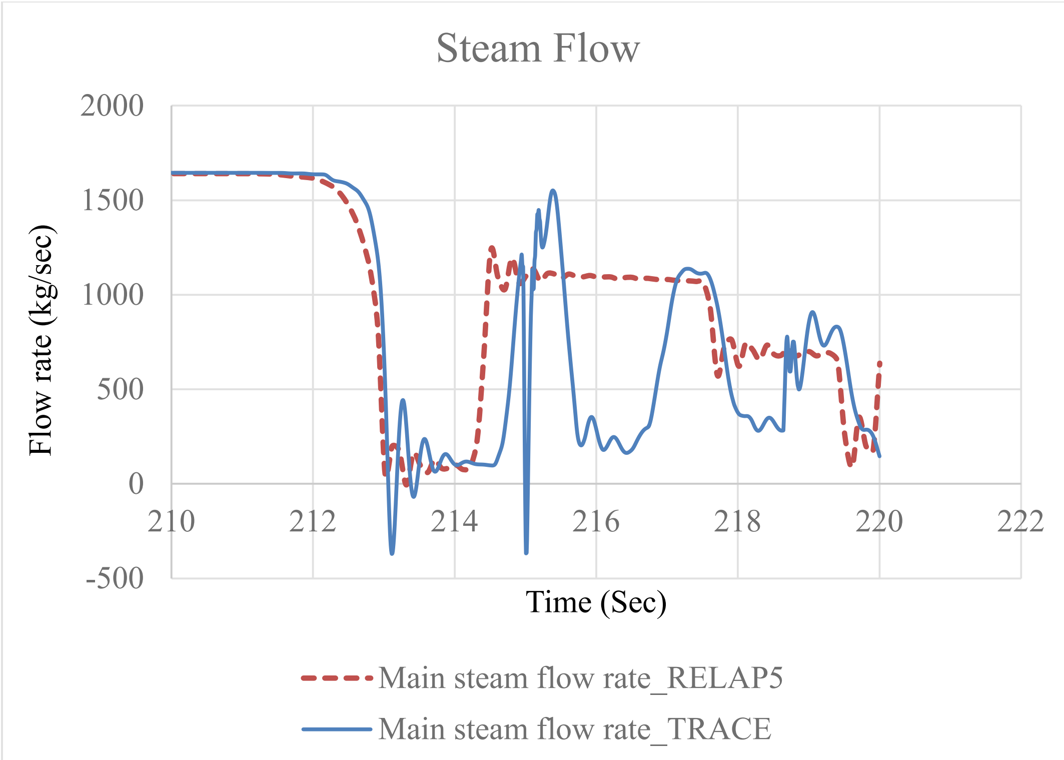

Figure 16 Steam flow variation during the MSIVC hypothetical accident

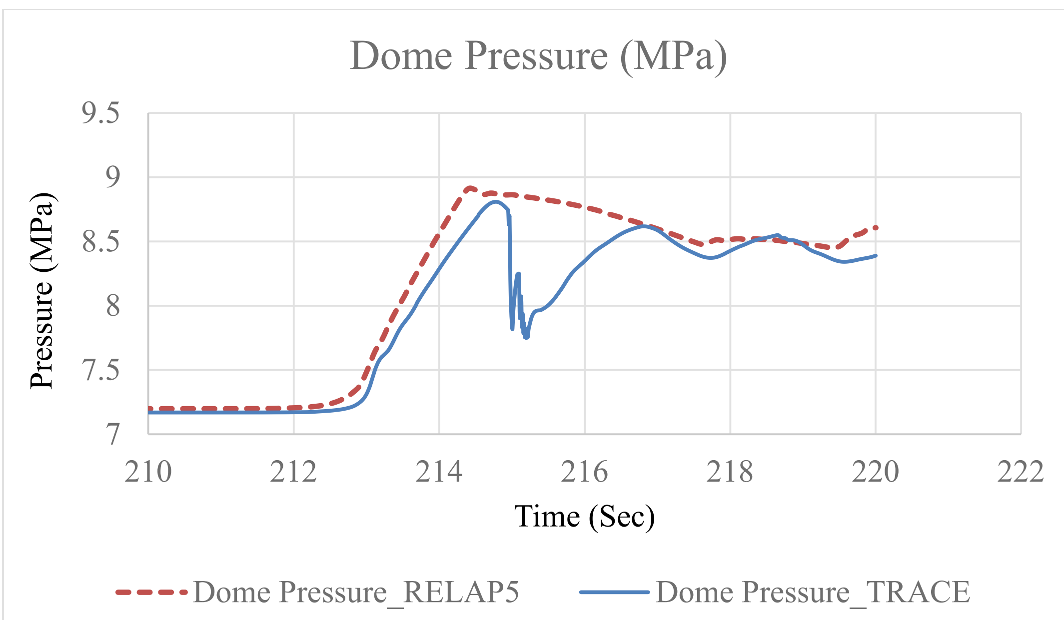

Figure 17 Dome pressure variation during the MSIVC hypothetical accident

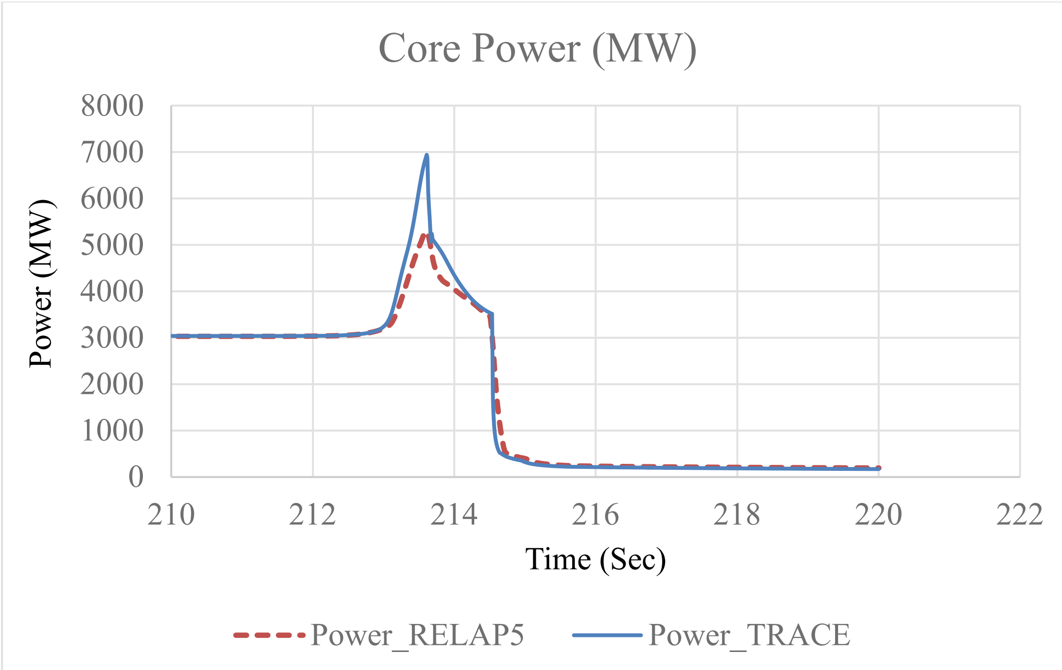

Figure 18 Core power variation during the MSIVC hypothetical accident

4.1.2 Fuel Rod Properties

Figure 19 shows the cladding temperature variation of single fuel rod which was divided into 12 nodes during the MSIVC transient. During the MSIVC transient, the cladding temperature increased because of the increasing power ratio. In this figure, it is obvious that the temperature of node 11 and 12 increased differently compared to temperature of other nodes. It is because that as the SRVs started to open at 214 seconds, the sudden pressure decreasing would cause the water flashing inside the reactor pressure vessel. The water flashing might influence the heat transfer efficiency of the top of fuel rods. To ensure this phenomenon, the RELAP5 analysis results were checked and it is found that heat transfer coefficient of the top of fuel rods is indeed decreased from 214 seconds to 218 seconds. In addition, the power ratio of node 12 is lower than that of node 11 so that the cladding temperature of node 12 did not increase so much as that of node 11.

Figure 20 shows the average cladding hoop strain variation during the MSIVC transient. Once the power and temperature increased, the gap gas would inflate and the fuel cladding would expand with heat, which was known as the thermal hoop strain. However, the increasing dome pressure would squeeze the fuel cladding which would cause negative elastic hoop strain, especially at the lower position such as Node 1. Based on the interaction of these two types of hoop strain, it can be determined that the cladding expanded or shrank during the transient. From Figure 4, it shows that most part of the fuel rod shrank but for the node 11 and 12, the cladding expanded because of the increasing cladding temperature. Nonetheless, even node 11 of fuel rod expanded obviously, the hoop strain is still not higher than the acceptance criteria 0.01. In addition to the cladding hoop strain, the fuel pellet enthalpy should also be noticed in overpressurization transient. Figure 21 shows the enthalpy variation during the transient. The peak value is about 250 kJ/kg (52.08 cal/g), which is much lower than the criteria 170 cal/g. From these three criteria mentioned above (temperature, hoop strain and enthalpy), it is known that the fuel rods would keep good integrity during the MSIVC transient.

In addition to obtaining the mechanical variation of fuel rod during the transient, comparing these analysis results with past research is another main goal of current research. Figure 22 shows the comparison of cladding temperature of RELAP5-FRAPTRAN 1.5 and TRACE-FRATRAN 1.4 analyses. To simplify the comparison, 12-node data is averaged. From this figure, it is obvious that at the steady state, the cladding temperature of RELAP5-FRAPTRAN 1.5 analysis is higher than that of TRACE-FRATRAN 1.4 analysis because the RELAP5 calculation predicted a higher coolant temperature which is an input condition for FRAPTRAN code. Further, the cladding temperature difference of steady state might also come from different FRATRAN code version. However, the influence of version should be checked with more research.

As the transient started and power increased, both cladding temperature increased. At 213.31 seconds, reactor scrammed and hence the fuel pellet stopped generating heat. Further, the power scrammed at 214 seconds, the SRVs opened and released vapor which would efficiently decreased the energy inside the reactor pressure vessel. For TRACE-FRAPTRAN 1.4 analysis, at about 215 seconds, the cladding temperature increased again because the SRVs returned to close and the main steam flow decreased so that the heat could not be carried with vapor until the SRVs opened and mains steam flow increased again at 216 seconds. On the contrary, SRVs of the RELAP5 model were kept open from 214 to 218 seconds. The steam flow kept releasing energy of the reactor core. Hence, the cladding temperature of RELAP5-FRAPTRAN analysis kept decreasing during the transient.

Figure 23 shows the difference of cladding hoop strain between RELAP5-FRAPTRAN 1.5 and TRACE-FRAPTRAN 1.4 analysis models. At about 213 seconds, the reactor scrammed and the gap gas pressure decreased as a result. However, the dome pressure started increased. Hence, the cladding hoop strain decreased from 213 seconds. After that, the cladding temperature started to increase at 214 seconds which means the thermal hoop strain would against the elastic hoop strain. The cladding hoop strain increased at 214 seconds for RELAP5-FRAPTRAN 1.5 model and at about 215 seconds for TRACE-FRAPTRAN 1.4 model. Then, the cladding temperature decreased and the elastic hoop strain (coolant pressure) dominated the total hoop strain. The cladding hoop strain kept decreasing to the end of the analysis. However, it is worth noticing that the cladding hoop strain of TRACE-FRAPTRAN 1.4 model vibrated from 215 to 220 seconds because the dome pressure from TRACE analysis varied in this duration. On the contrast, the dome pressure from the RELAP5 analysis decreased with a smooth trend. As a result, the cladding hoop strain after SRVs open of these two models is slightly different.

The last compared data is fuel enthalpy. With a higher coolant temperature, temperature of the fuel pellet of RELAP5-FRATRAN model is also higher than that of TRACE-FRAPTRAN model. Hence, the fuel enthalpy of RELAP5-FRATRAN model is higher than that of TRACE-FRAPTRAN model. However, from Figure 24, it can be noticed that the enthalpy curves of these two models come closer at the end of transient because the cladding temperature of RELAP5-FRATRAN model decreased more than that of TRACE-FRAPTRAN model, which implies that the fuel temperature of RELAP5-FRAPTRNA model reached to a lower value. The fuel enthalpy decreased as a result.

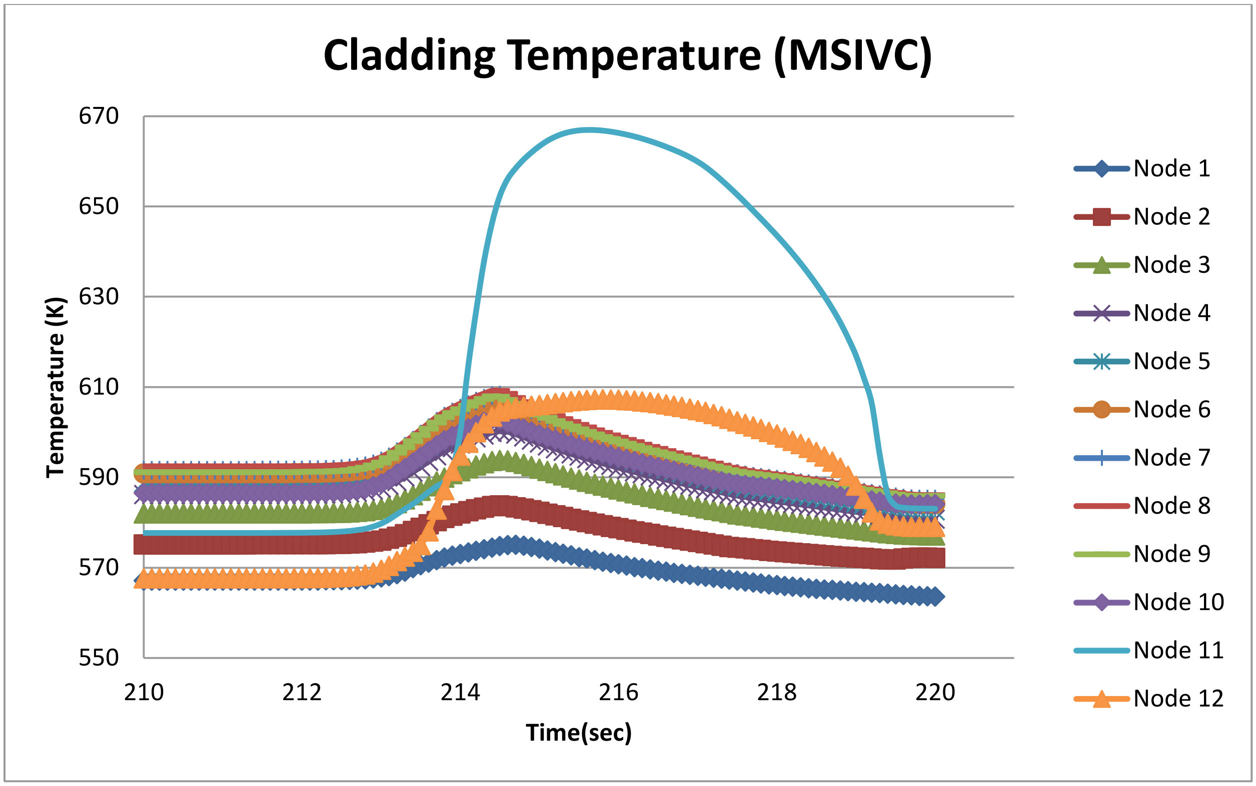

Figure 19 Cladding temperature of MSIVC transient

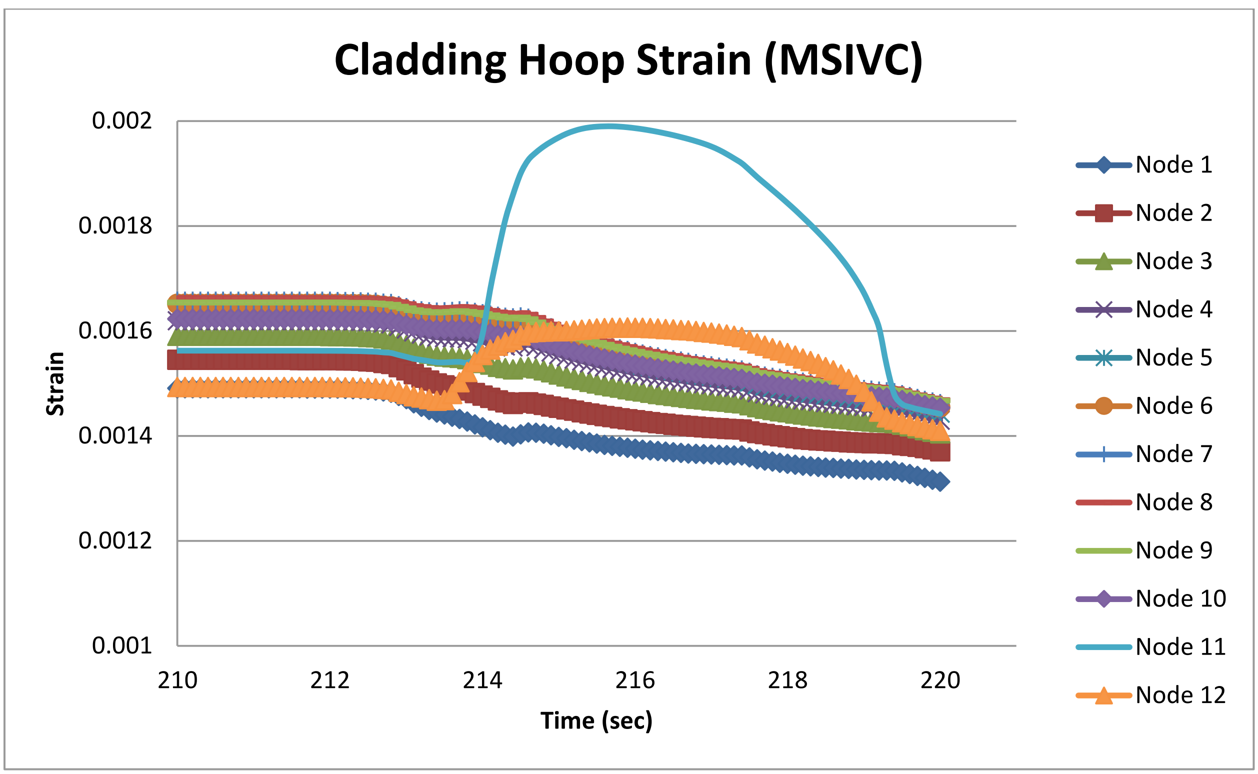

Figure 20 Cladding hoop strain of MSIVC transient

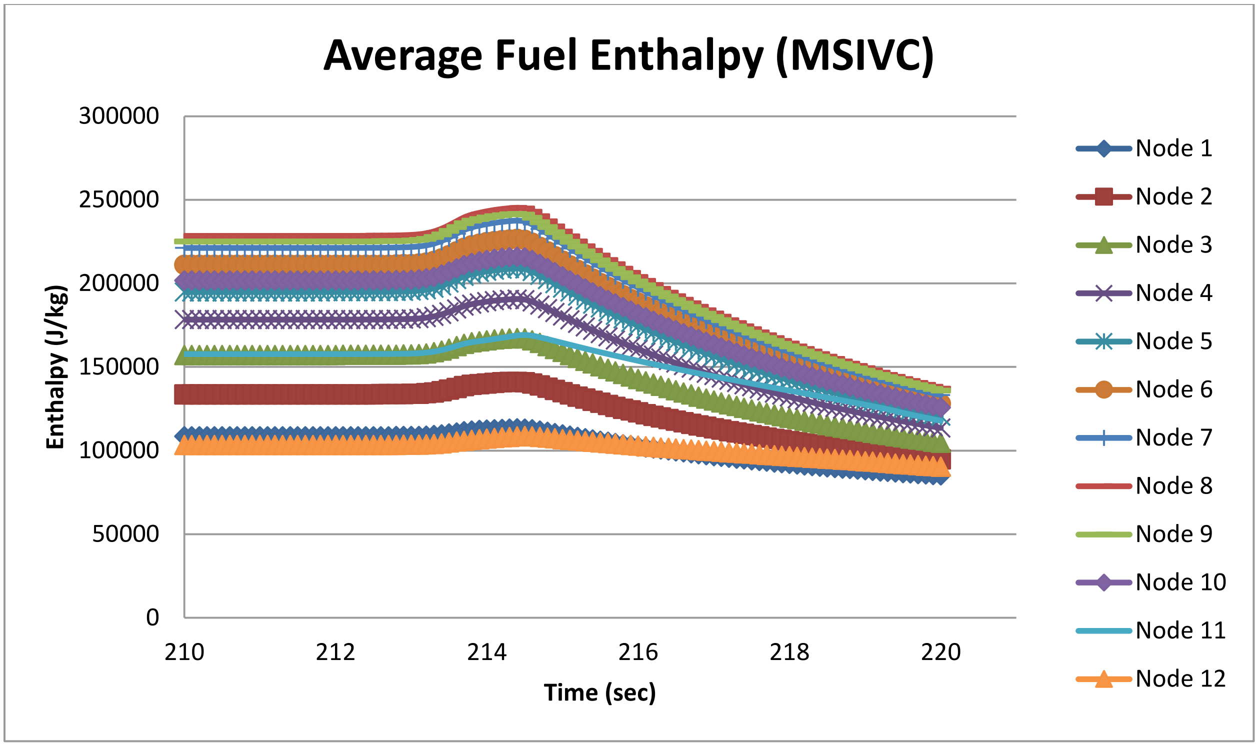

Figure 21 Fuel enthalpy of MSIVC transient

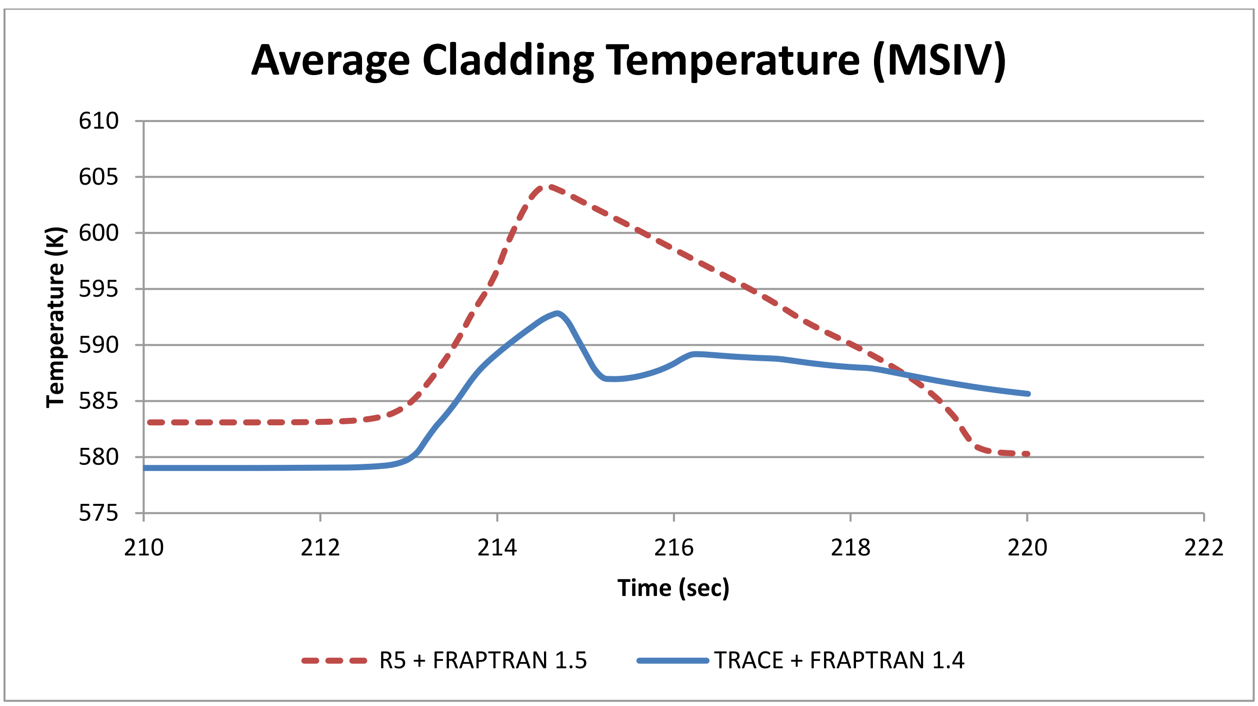

Figure 22 Cladding temperature comparison of MSIVC transient

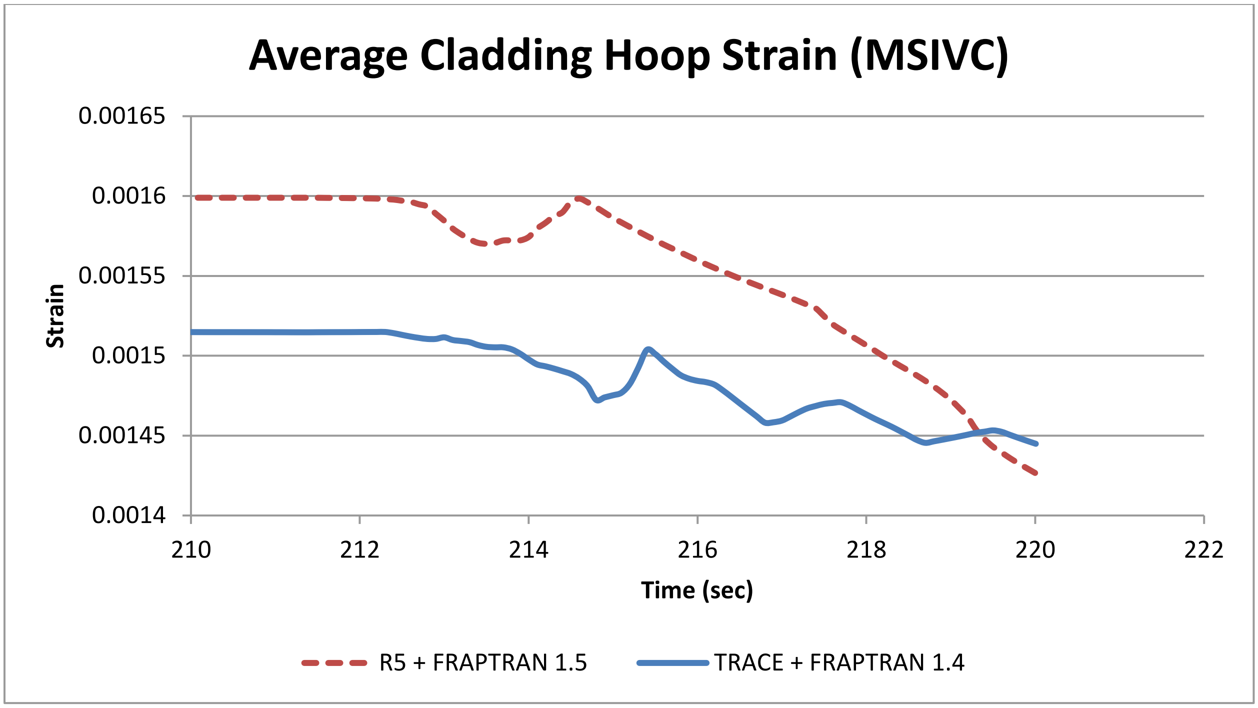

Figure 23 Cladding hoop strain comparison of MSIVC transient

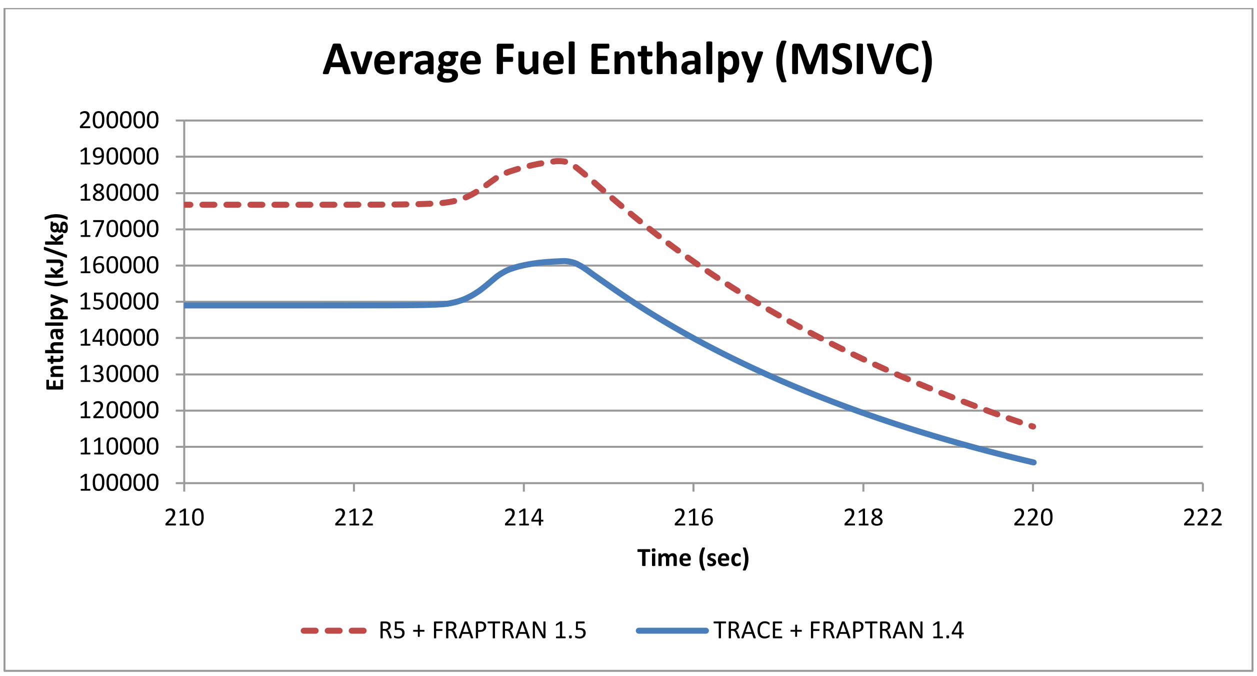

Figure 24 Fuel enthalpy comparison of MSIVC transient

4.2 Turbine Trip with Bypass Failure

4.2.1 Thermal Hydraulic Analysis data results

The TSVs started to close at 500 second and fully closed at 500.1 second. Figure 25 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 26. 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 27. 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 27, 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 26 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 26, 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 25. 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.

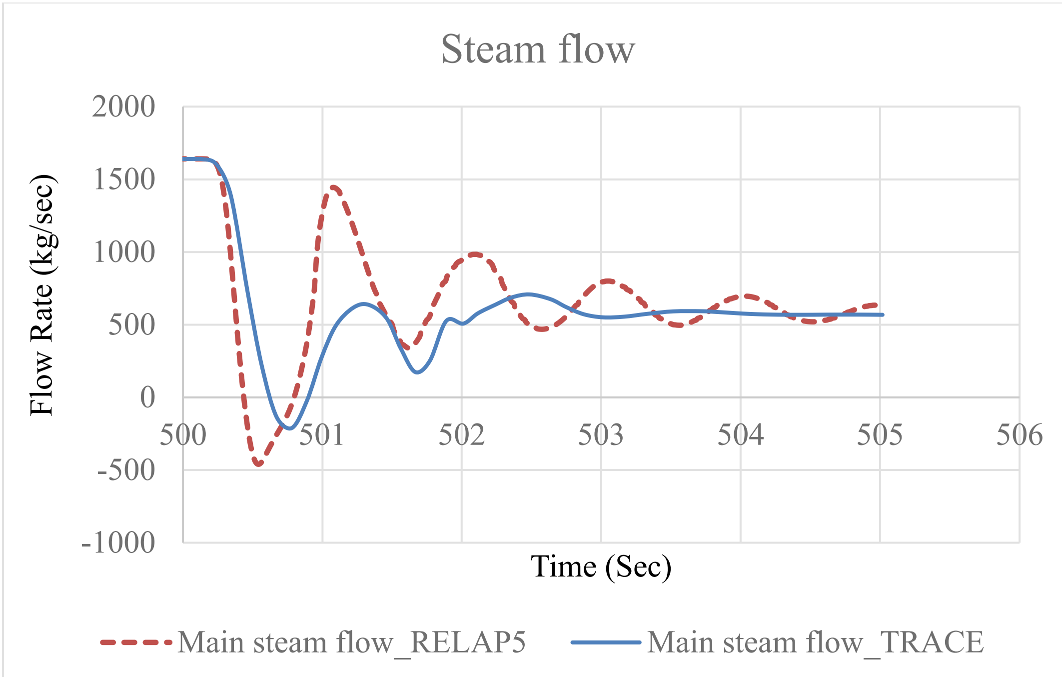

Figure 25 Steam flow variation during the TTBF hypothetical accident

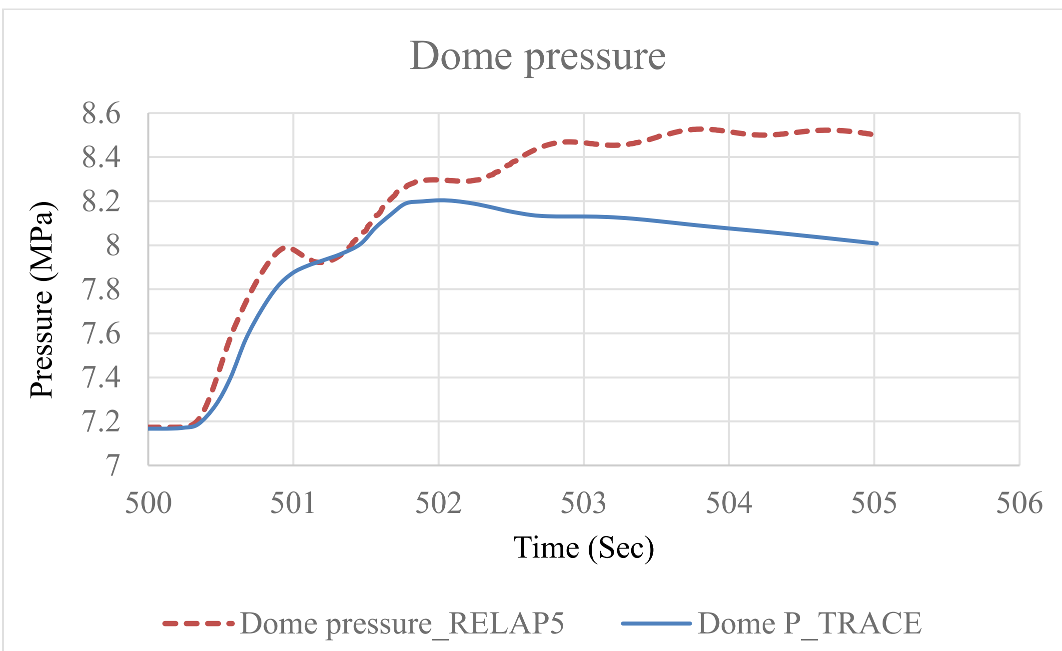

Figure 26 Dome pressure variation during the TTBF hypothetical accident

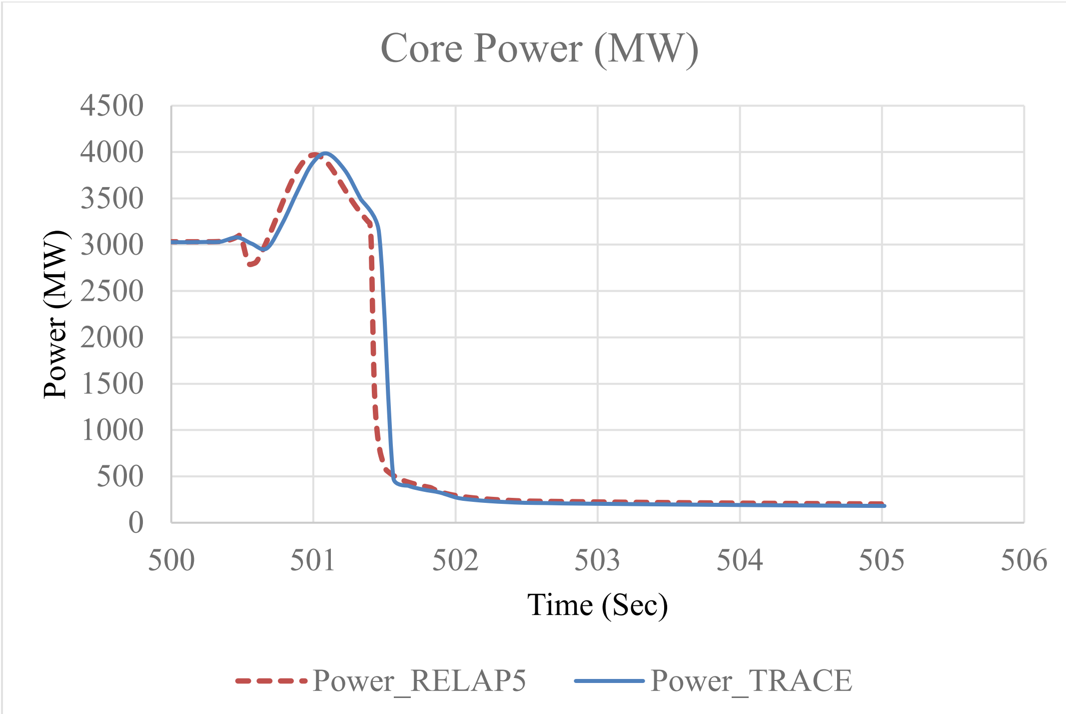

Figure 27 Core power variation during the TTBF hypothetical accident

4.2.2 Fuel Rod Properties

As shown in Figure 28, the cladding temperature increased with power increasing. From this figure, it is obvious that the cladding temperature is far away from the criterion 1088K. Due to the increasing temperature of the fuel rod, the gas inside would be inflated. As a result, the gap gas pressure inside cladding would expand the cladding hoop strain. Further, the increasing cladding temperature would also cause the thermal cladding hoop strain expansion. The cladding hoop strain would increase with increasing temperature. However, at the beginning of the transient (500.3 to 500.5 seconds), the influence of dome pressure increasing is much more than that of fuel rod temperature. The dome pressure dominated the cladding deformation. After that, the influence of temperature became large and as a result the cladding hoop strain increased until 501.2 seconds, the time step of reactor scram. After the reactor scram, core power dropped rapidly and the temperature decreased. The dome pressure outside cladding then dominated the hoop strain variation again. During the transient, the cladding hoop strain was not larger than criterion 0.01 (as shown in Figure 29). In addition to the cladding temperature and hoop strain, fuel pellet enthalpy is also an important criterion during the transient. Figure 30 shows that the maximum enthalpy in the TTBF transient of 12 nodes of fuel rod is about 231800 J/kg, which was much lower than the criteria 710600 J/kg.

Same as the MSIVC case, the fuel rod properties are compared to past research. As mentioned in 4.1 section, the variation of 12 positions on the fuel rod was averaged to make the figure clear. Figure 31 is the cladding temperature comparison of RELAP5-FRAPTRNA 1.5 and TRACE-FRAPTRAN 1.4 analysis. From this figure, it shows that the cladding temperature variation of these two analysis was quiet similar. The temperature of RELAP5-FRAPTRAN 1.5 case increased slightly earlier than that of TRACE-FRAPTRAN 1.4 case. It is because the power of

RELAP5 analysis increased earlier as shown in Figure 41. Figure 32 shows that the cladding hoop strain of these two cases is similar but the strain of the RELAP5-FRAPTRNA 1.5 case is lower at the end of the transient. It is reasonable because the dome pressure of RELAP5 analysis kept in a higher value than that of TRACE analysis. Figure 33 shows the fuel enthalpy comparison. For both analysis, the enthalpy variation and value are quiet similar. Both curves in Figure 33 decreased slowly due to the power scram.

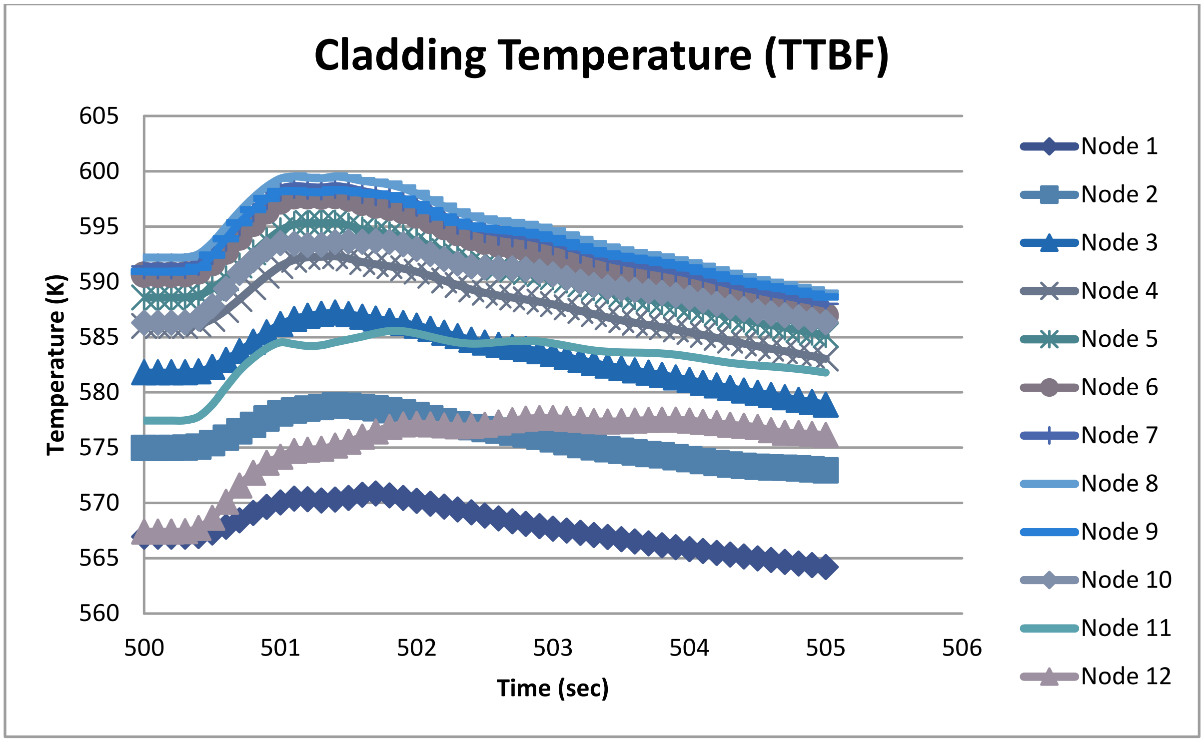

Figure 28 Cladding temperature of TTBF transient

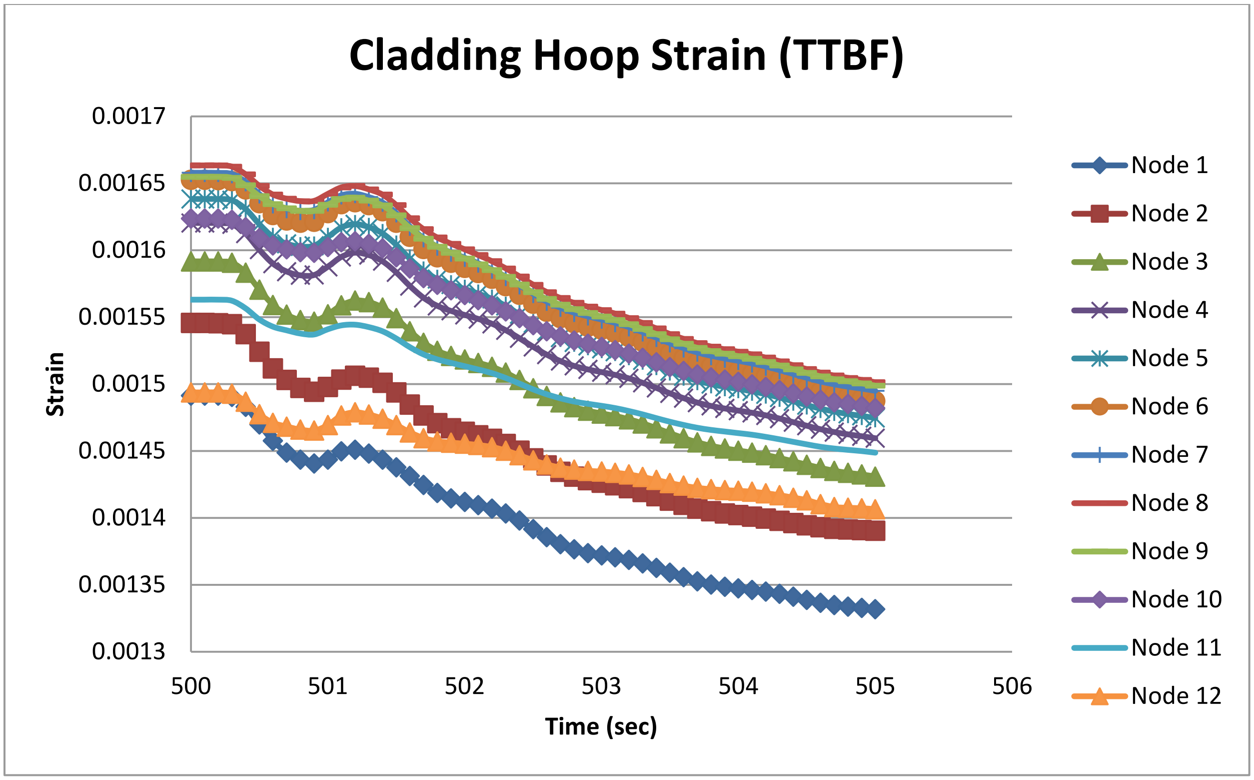

Figure 29 Cladding hoop strain of TTBF transient

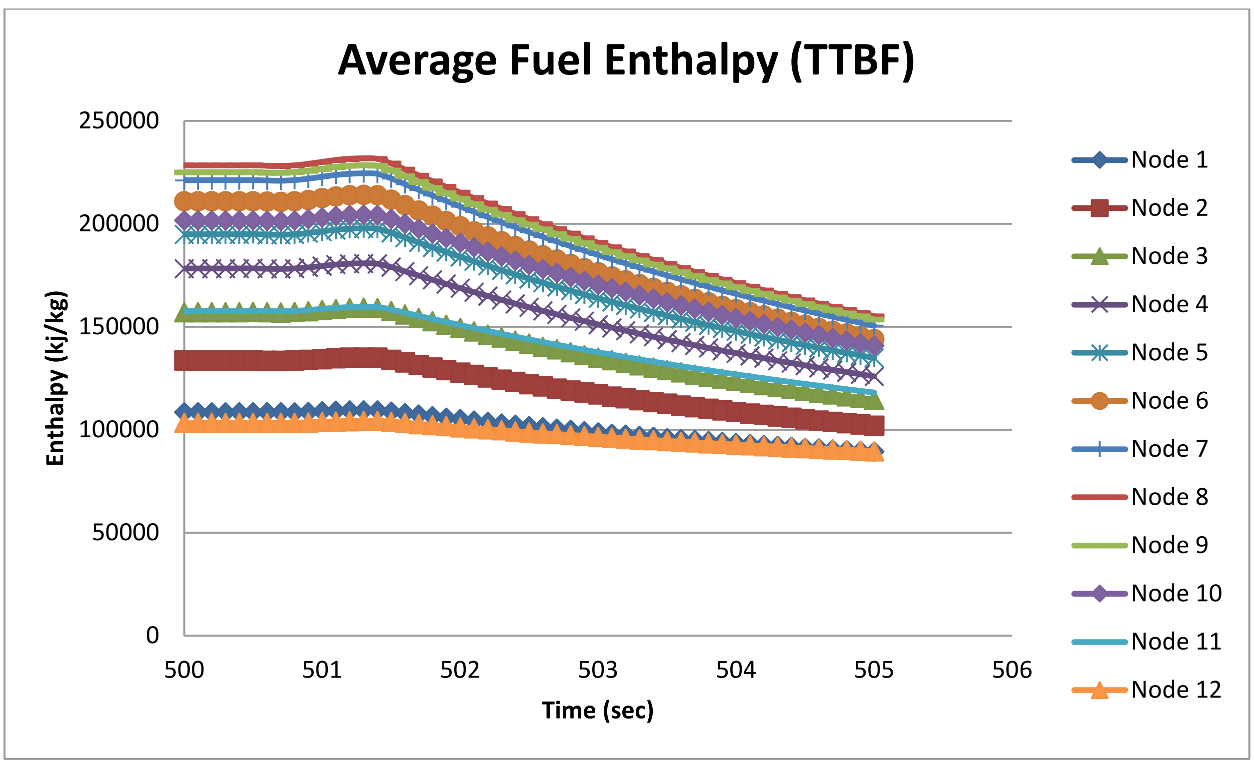

Figure 30 Fuel enthalpy of TTBF transient

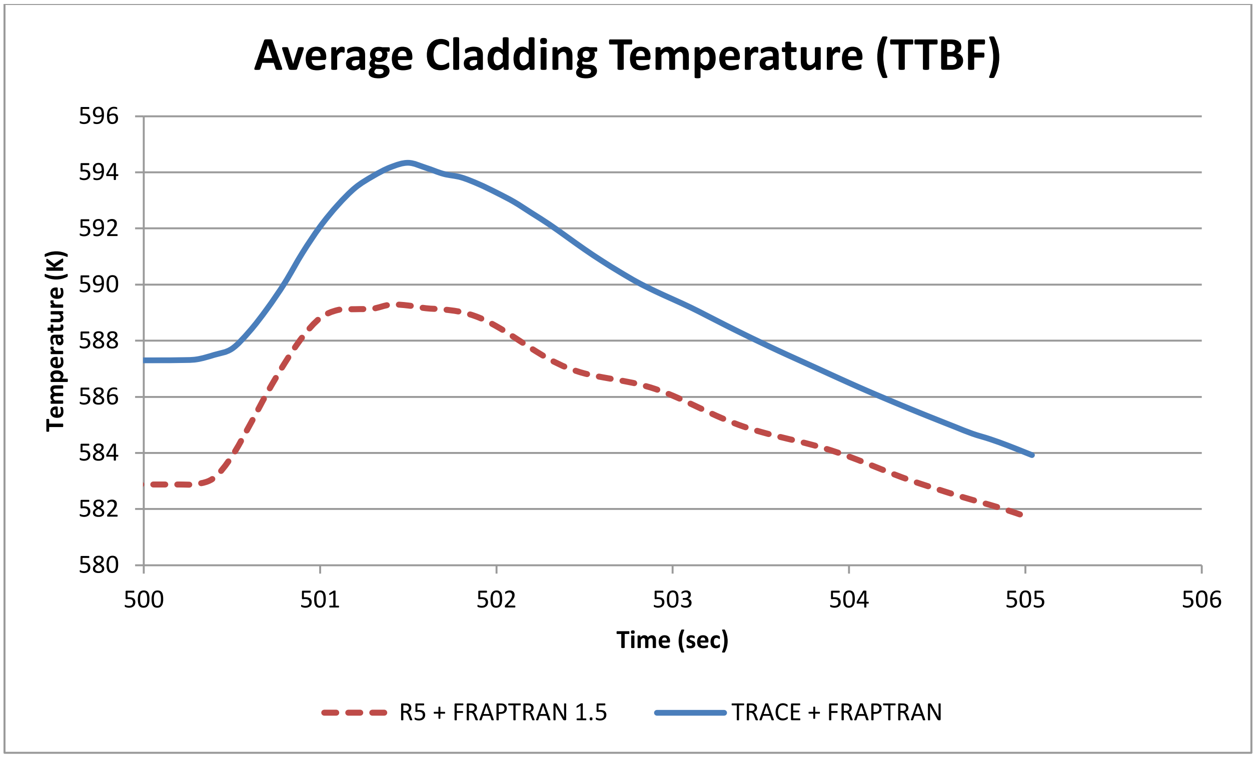

Figure 31 Cladding temperature comparison of TTBF transient

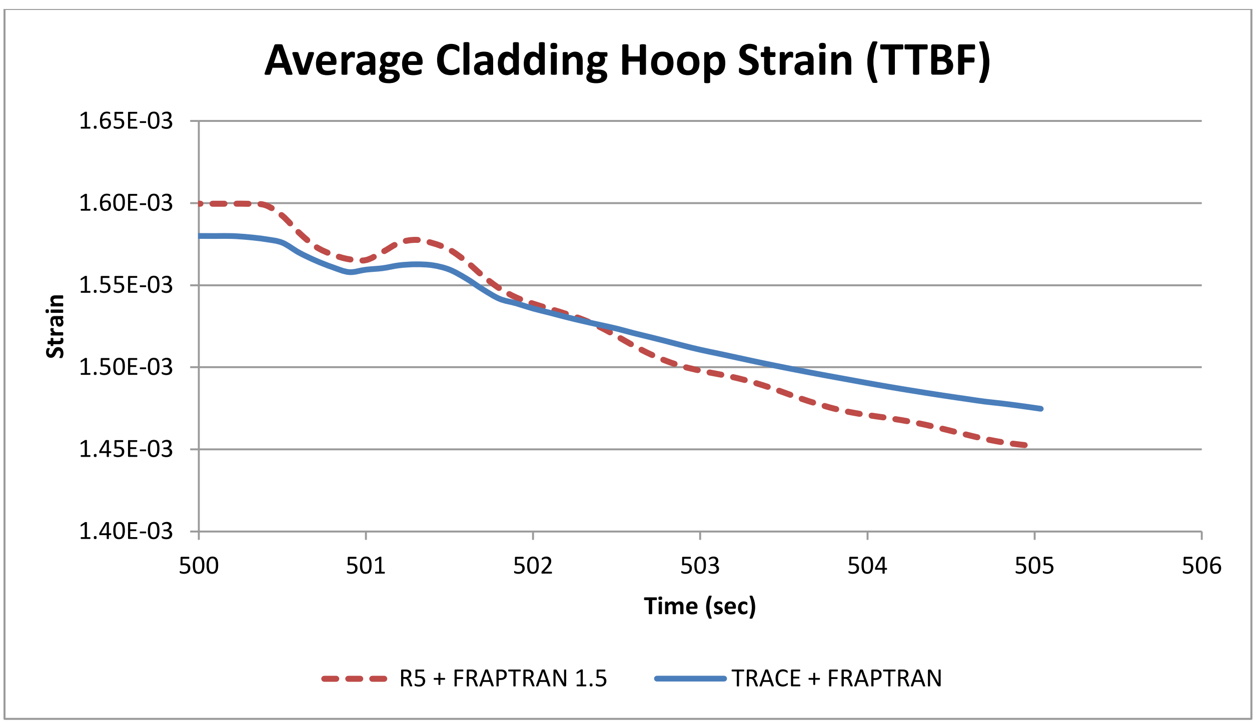

Figure 32 Cladding hoop strain comparison of TTBF transient

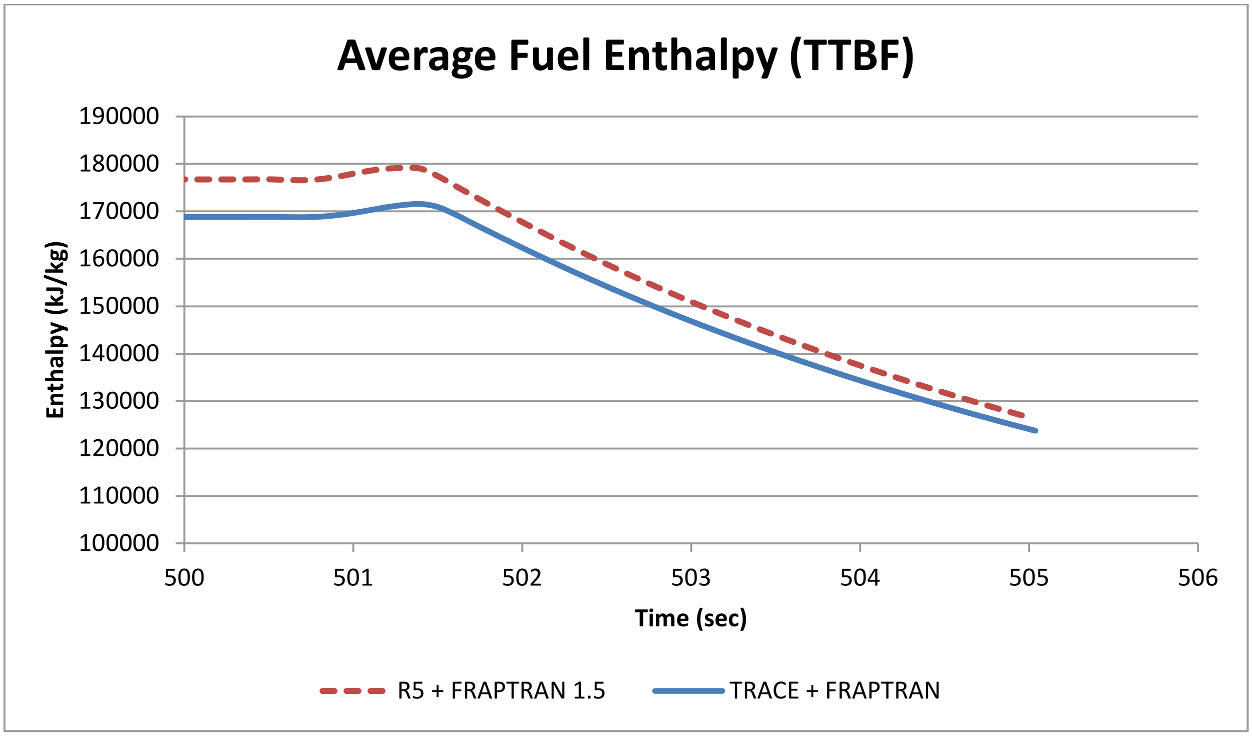

Figure 33 Fuel enthalpy comparison of TTBF transient

4.3 Load Rejection with Bypass Failure

4.3.1 Thermal Hydraulic Analysis data results

Figure 34 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 35. As a result, the void fraction inside the reactor core decreased which caused a positive reactivity feedback. The power increased as shown in Figure 36. Different from previous two cases, the reactor scram signal was generated at once as the turbine control valves closed. The power ratio 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 ratio. However, at 211.1 second for RELAP5 model and 211.2 second for TRACE model, the negative scram reactivity feedback dominated the power ration 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 pressures 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 37 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. The Kuosheng NPP was under control in the LRBF transient.

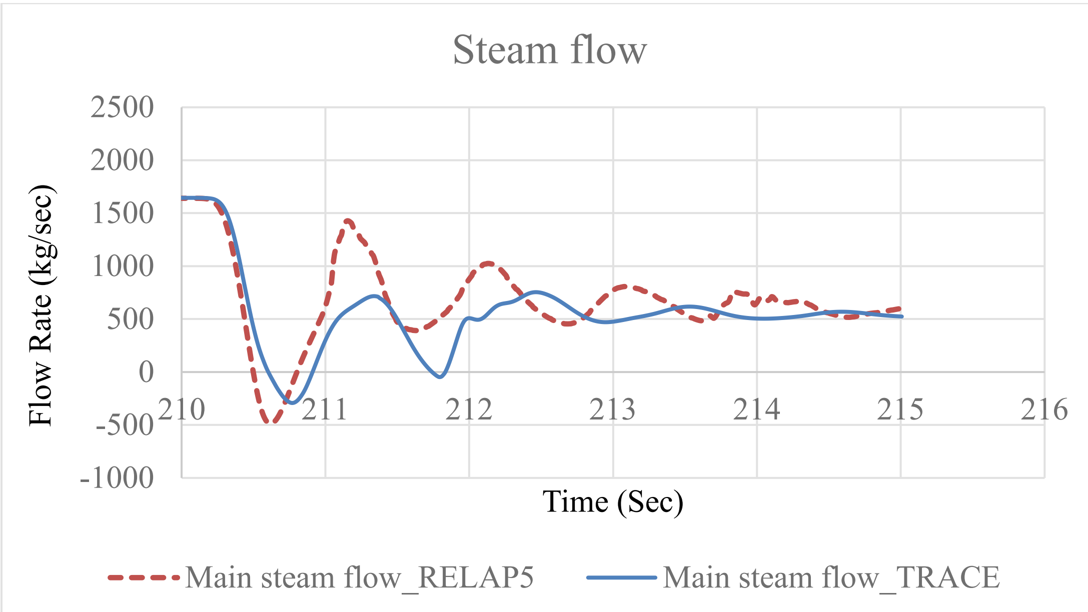

Figure 34 Steam flow variation during the LRBF hypothetical accident

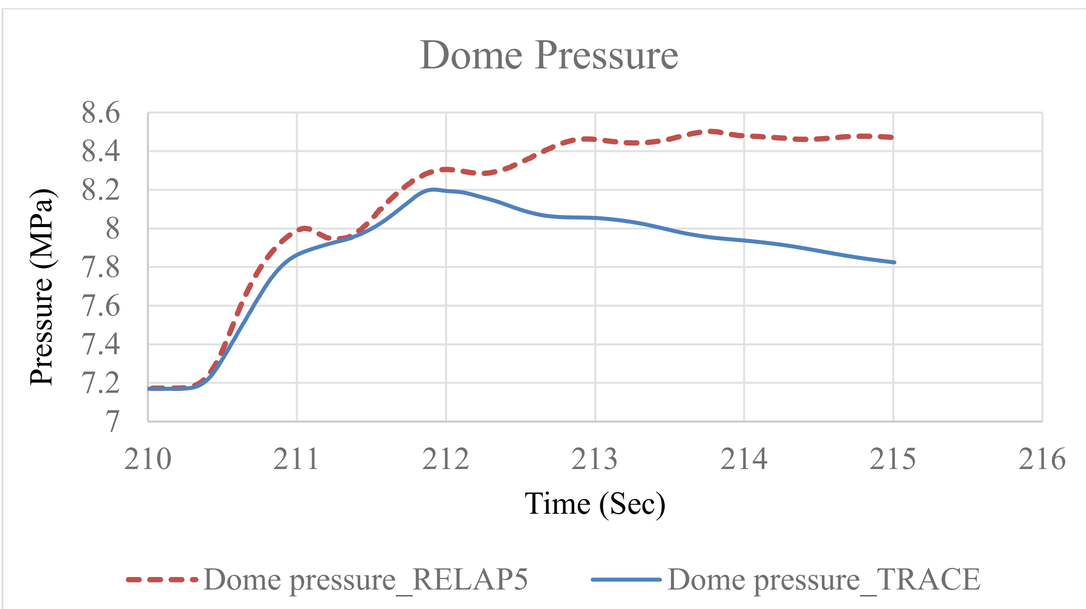

Figure 35 Dome pressure variation during the LRBF hypothetical accident

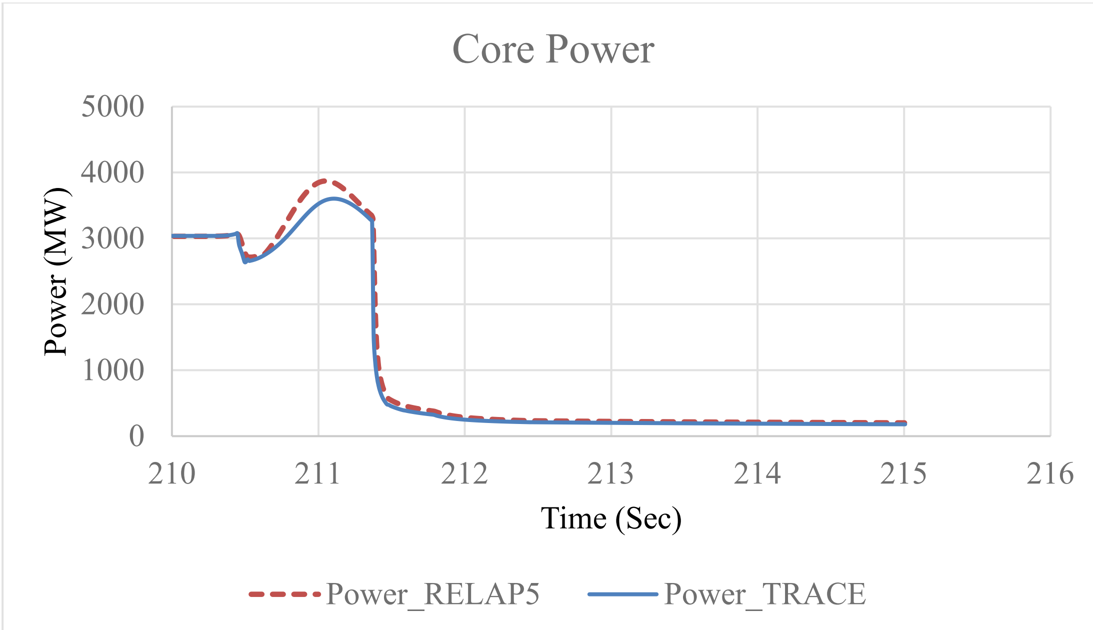

Figure 36 Core power variation during the LRBF hypothetical accident

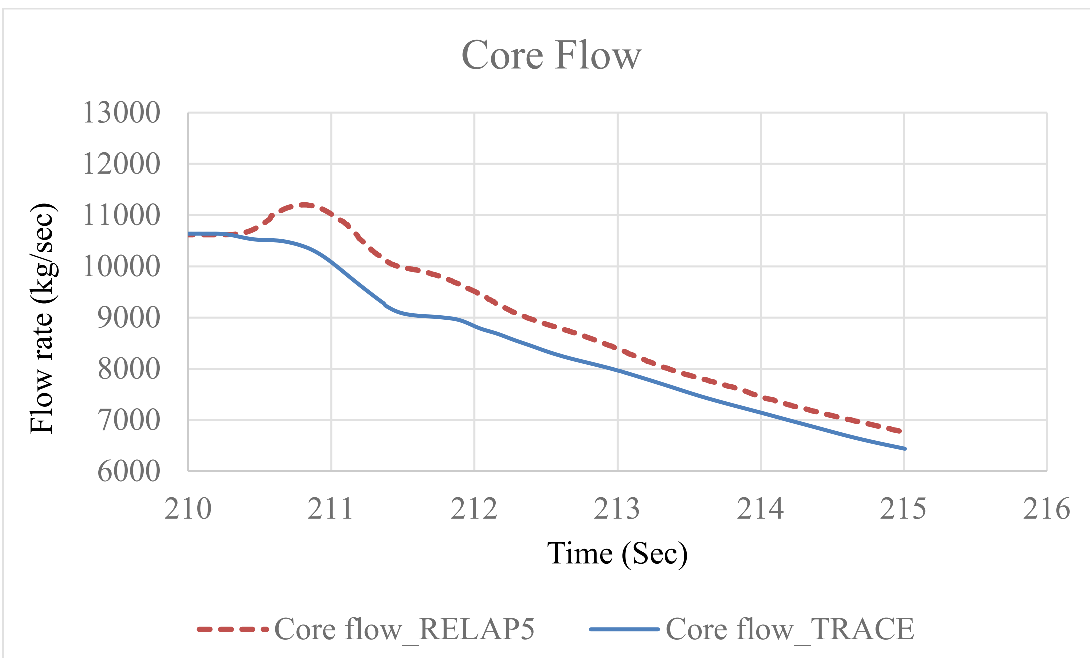

Figure 37 Core flow variation during the LRBF hypothetical accident

4.3.2 Fuel Rod Properties

Once the TCVs started to close, the power ratio increased and as a result, the cladding temperature of all positions increased as shown in Figure 38. Theoretically, the increasing power and temperature would inflate the gap gas and cause the thermal hoop strain which expanded the cladding hoop strain. However, the TCVs closed in a short time and as a result the dome pressure increasing and elastic hoop strain would dominate the cladding deformation at the beginning of the transient. Hence, as shown in Figure 39, the cladding hoop strain decreased at the beginning of transient until 211 seconds. At about 211.1 seconds, the temperature was high enough to dominate the cladding deformation. However, the temperature expanded the cladding just for 0.2 to 0.3 second because the reactor scrammed and hence the fuel and cladding temperature decreased. The dome pressure squeezed the cladding again so that the cladding hoop strain kept decreasing to the end of analysis. From the figure, it is obvious that cladding hoop strain of LRBF transient was not larger than criteria 0.01. In addition to the cladding temperature and hoop strain, fuel enthalpy is another important criterion for fuel rod analysis. From Figure 40, maximum of the fuel enthalpy for all positions is just 230750 J/kg, which is far from the acceptance limit 710600 J/kg.

In addition to obtaining the mechanical variation of fuel rod during the transient, comparing these analysis results with past research is another main goal of current research. Figure 41 shows the comparison of cladding temperature of RELAP5-FRAPTRAN 1.5 and TRACE-FRATRAN 1.4 analyses. To simplify the comparison, 12-node data is averaged. Same as comparisons of the other two cases, as shown in Figure 42, it is obvious that at the steady state, the cladding temperature of RELAP5-FRAPTRAN 1.5 analysis is higher than that of TRACE-FRATRAN 1.4 analysis because the RELAP5 calculation predicted a higher coolant temperature which is an input condition for FRAPTRAN code. In addition, for the cladding hoop strain comparison shown in Figure 42, the variation predicted by RELAP5-FRAPTRAN 1.5 model is higher than that of TRACE-FRATRAN 1.4 model due to the cladding temperature difference. Further, in this figure, a peak value of RELAP5-FRAPTRAN 1.5 variation was found at about 211.3 seconds because the dome pressure of RELAP5 analysis was in an obvious decreasing trend during 210.98 to 211.2 seconds so that the cladding expanded in this period. After that, variations of both models are quiet similar. However, the cladding hoop strain obtained from RELAP5-FRAPTRAN 1.5 model decreased more than that obtained from TRACE-FRATRAN 1.4 model because the dome pressure of RELAP5 analysis kept in a higher value while the dome pressure of TRACE analysis decreased.

As for the fuel enthalpy, the RELAP5-FRAPTRAN 1.5 model predicted a higher value than TRACE-FRATRAN 1.4 model predicted as shown in Figure 43. The main reason of this difference would also be the temperature. Though the RELAP5-FRAPTRAN 1.5 model predicted a higher enthalpy, in general, the variation is quiet similar than that TRACE-FRAPTRAN 1.4 model predicted.

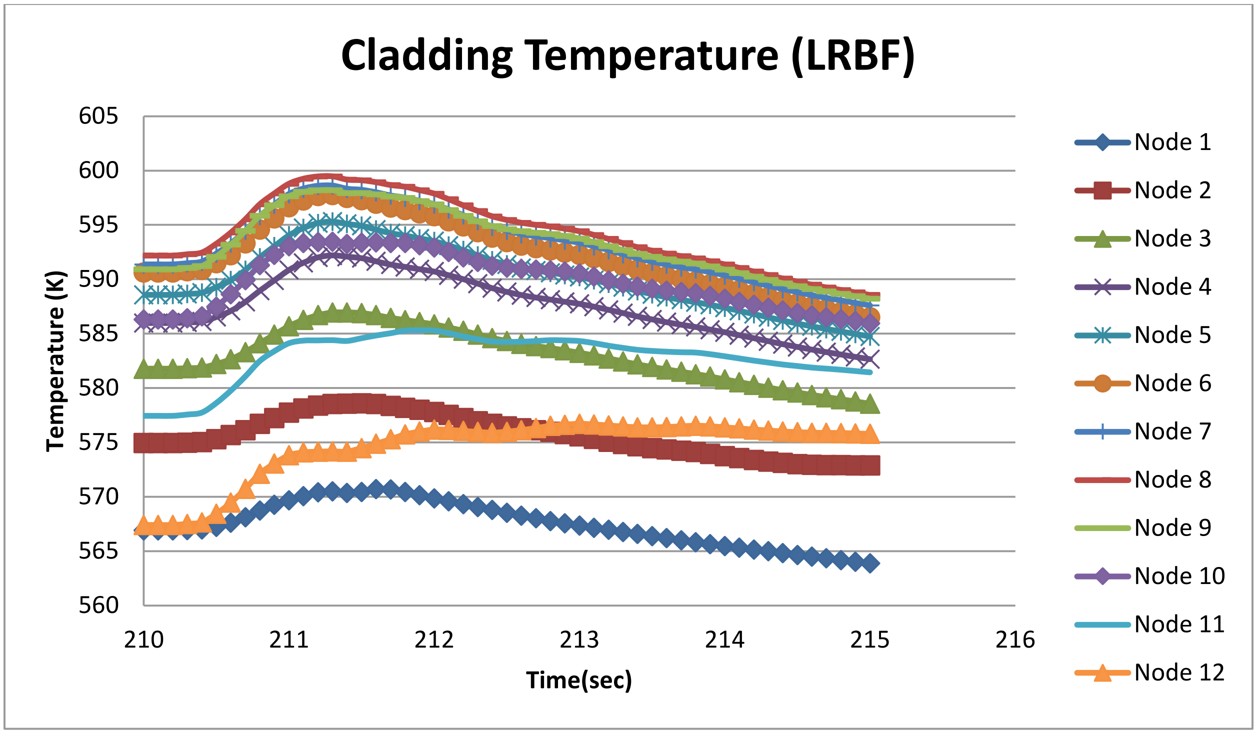

Figure 38 Cladding temperature of LRBF transient

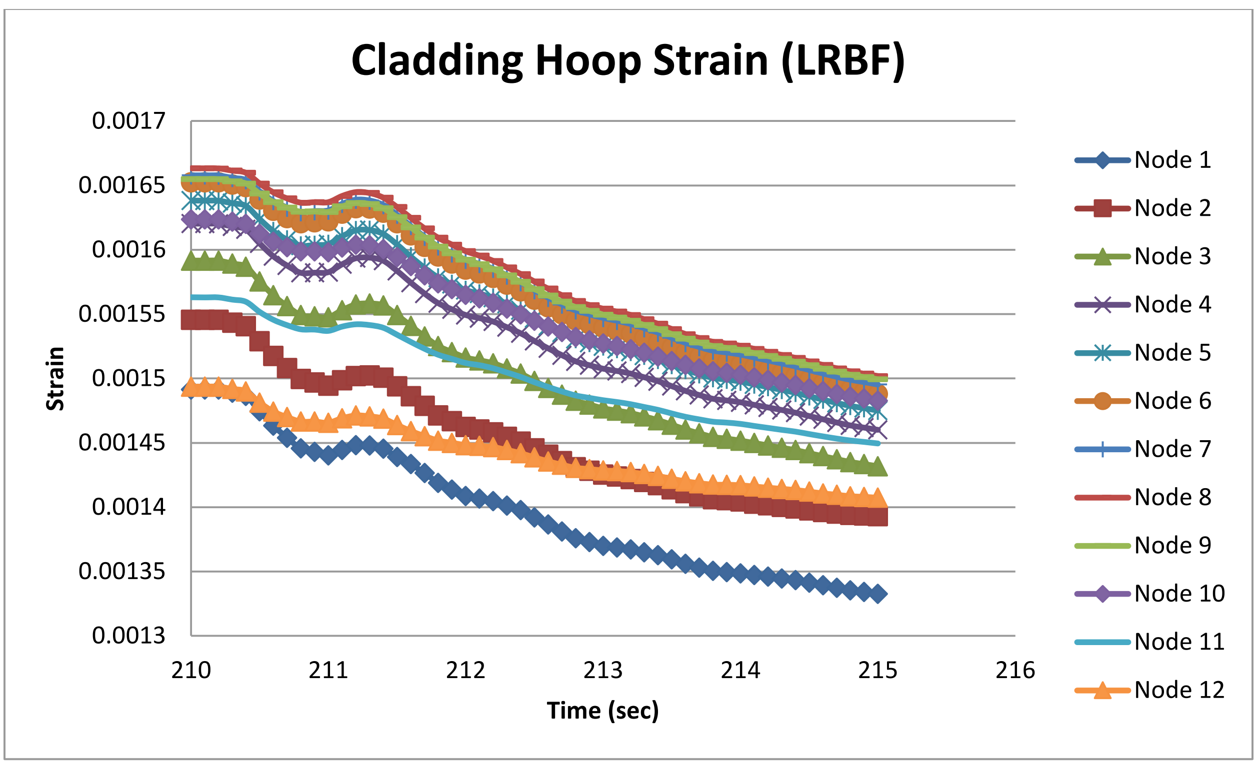

Figure 39 Cladding hoop strain of LRBF transient

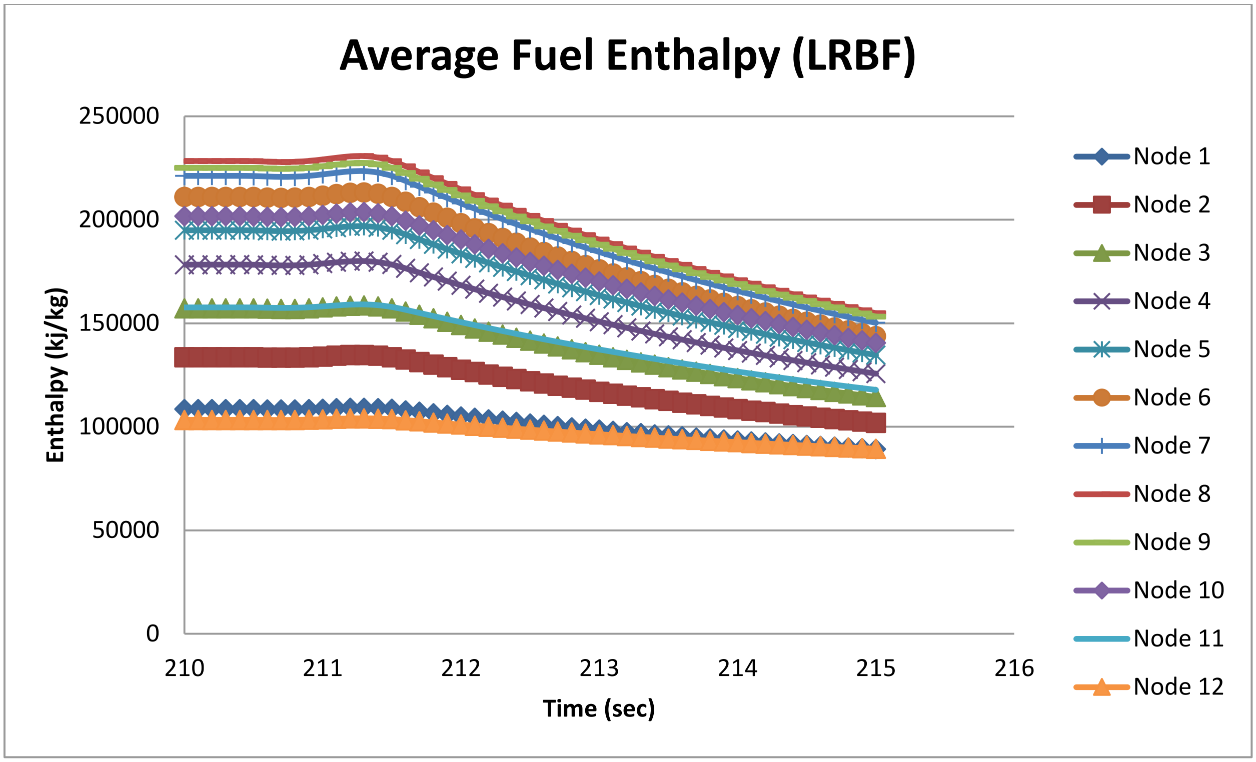

Figure 40 Fuel enthalpy of LRBF transient

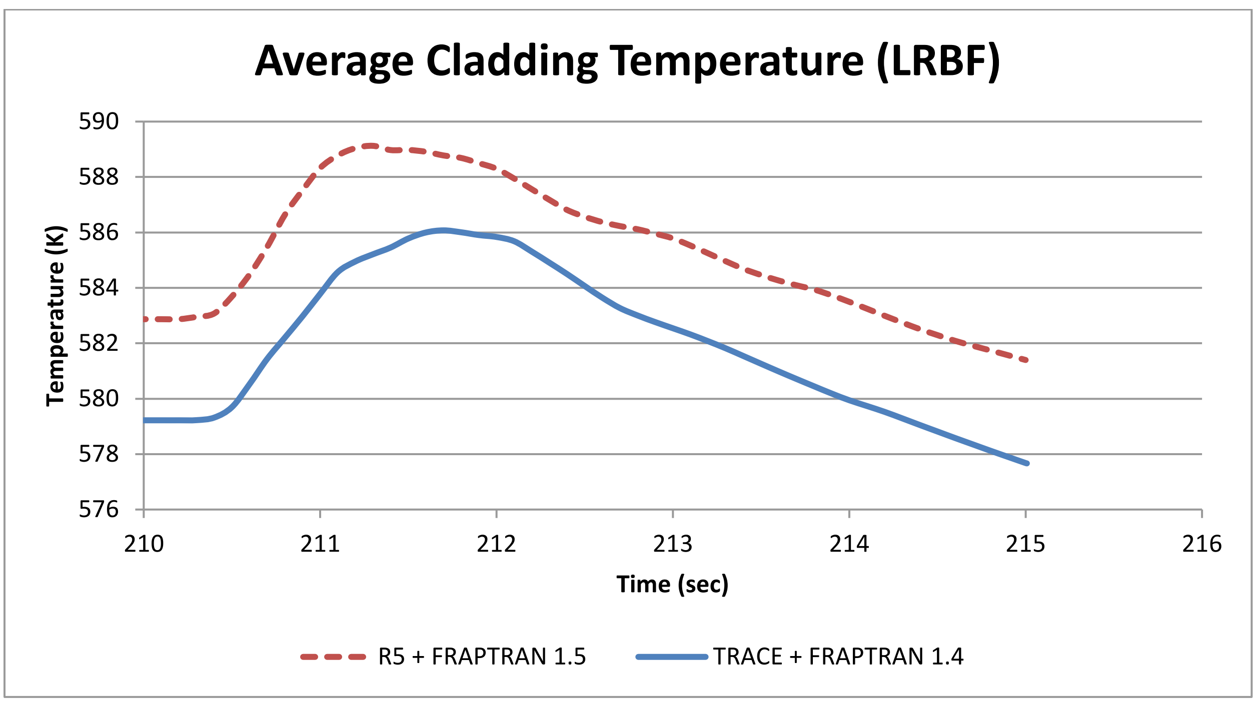

Figure 41 Cladding temperature comparison of TTBF transient

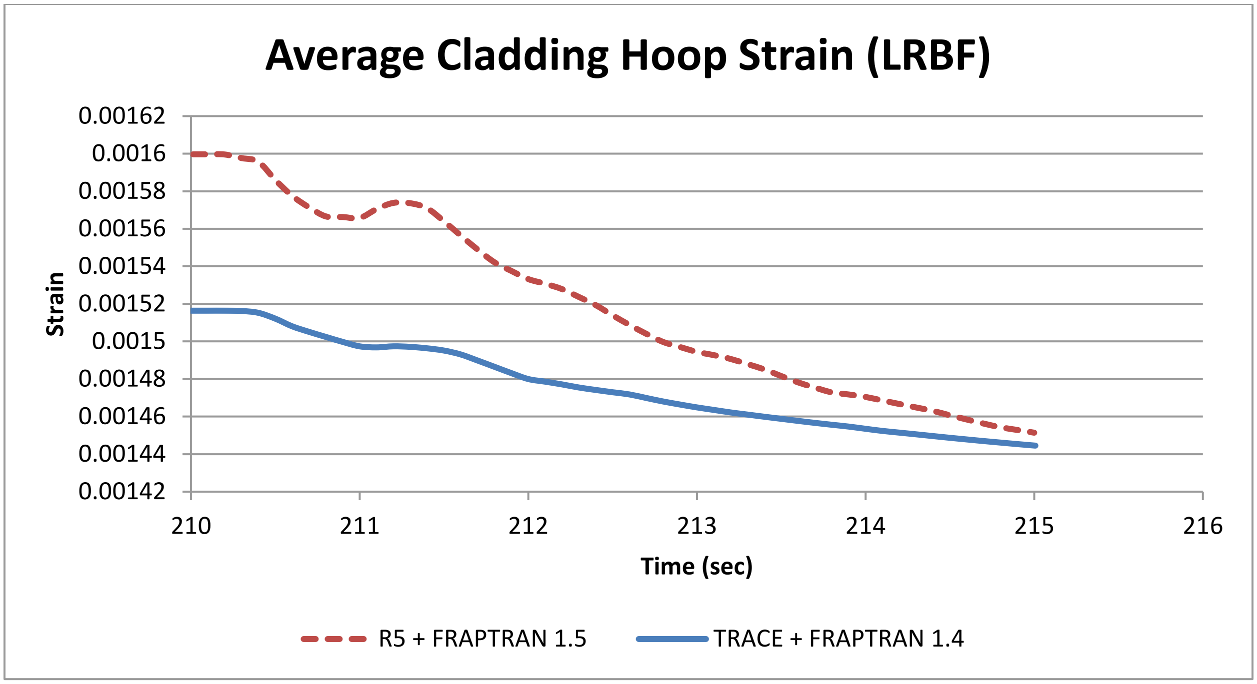

Figure 42 Cladding hoop strain comparison of LRBF transient

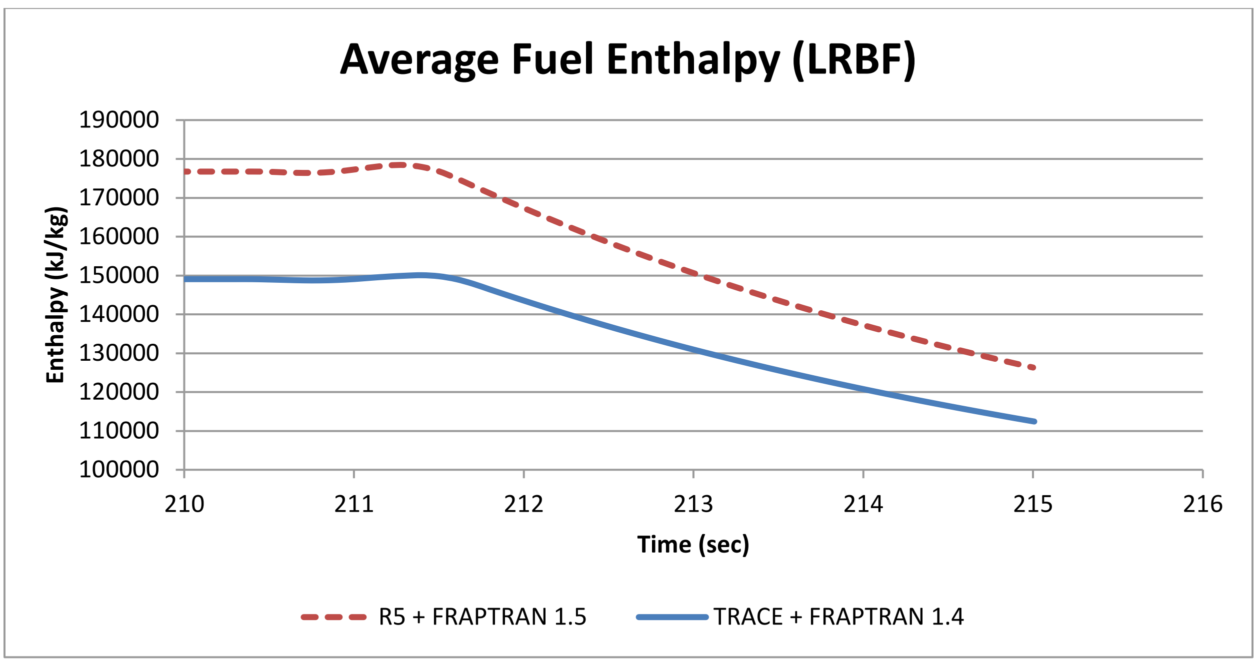

Figure 43 Fuel enthalpy comparison of LRBF transient

5

6 CONCLUSIONS

Before this research, the RELAP5/MOD 3.3 thermal hydraulic model and the TRACE-FRAPTRAN combination model for Kuosheng NPP have been developed. However, for the RELAP5 model, the fuel rods mechanical response during the transient cannot be determined. As for the TRACE-FRAPTRAN combination model, the manual data transferring process is quiet complicate. Hence, a RELAP5/MOD 3.3 and FRAPTRAN 1.5 combination model of Kuosheng NPP which can transfer the data results of RELAP5 code into FRAPTRAN input deck automatically was successfully developed with SNAP interface. To verify the capability of this model, three overpressurization hypothetical transient are performed. From the analysis data, it is known that the fuel rod properties would not exceed the criteria among these three cases. Further, the data results are also compared to the past research which was done by TRACE-FRAPTRAN model. From the comparisons, it is known that in general, fuel rod mechanical response of these two models are quiet similar but the RELAP5-FRAPTRAN 1.5 predicted higher values because of the higher coolant temperature from RELAP5 analysis results. This research also verified the applicability of AptPlot and python script job stream. With proper code descripts and settlement, the job streams of SNAP interface can save many human resources and time to perform analysis. With developing the thermal hydraulic and fuel rod mechanical combination model, the FRAPTRAN 1.5 is recommended because it allows more boundary condition data sets so that the pressure, power, heat transfer coefficient and coolant temperature from thermal hydraulic analysis need just few data reduction to meet the FRAPTRAN code requirements for boundary conditions. With the useful python scripts, thermal hydraulic and fuel rod mechanism combination analysis such as RELAP5-FRAPTRAN can be easily applied. In the future, researchers may apply this job stream on uncertainty analysis and determine how the thermal hydraulic uncertainties terms affect the mechanical response of fuel rods.

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10. SUPPLEMENTARY NOTES

K. Tien

11. ABSTRACT (200 words or less)

After the measurement uncertainty recapture power uprates, Kuosheng nuclear power plant (NPP) was uprated the power from 2894 MWt to 2943 MWt. For this 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 analyses should be done for ensuring that the power plant could maintain stability in higher power operating conditions. In this research, three overpressurization transients scenario including main steam isolation valves closure, turbine trip with bypass failure and load rejection with bypass failure would be performed by RELAP5 MOD3.3 code. Further, the thermal hydraulic properties of the reactor core will be transferred as the boundary conditions of FRAPTRAN code. With the boundary conditions from RELAP5 code, the fuel rod mechanical properties during the transient could be determined. In this research, the SNAP interface is applied so that the transferring process between RELAP5 and FRAPTRAN code can be completed automatically with Python Job Stream. That is, the researchers need not calculate and transfer the thermal hydraulic boundary conditions for FRAPTRAN analysis manually.

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