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International Agreement Report

Fuel Rod Performance Uncertainty Analysis During Overpressurization Transient for Kuosheng Nuclear Power Plant with TRACE/ FRAPTRAN/ DAKOTA Codes in SNAP Interface

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ABSTRACT

After the MUR (measurement uncertainty recapture) power uprates, Kuosheng Nuclear Power Plant has uprated the power to 2943 MWt. Recently, Taiwan Power Company is concerned in SPU (stretch power uprated) plan and uprates the power to 3030 MWt, which is 104.7% of the designed power. Before the stretch power uprates, several transient analysis should be done for ensuring that the power plant could maintain stability in high power operating conditions. The advanced thermal hydraulic analysis code TRACE which is conducted by U.S. NRC was applied for overpressurization transient including main steam line isolation valves closure, turbine stop valves closure and turbine control valves closure. This closure of valves increased the dome pressure; as a result, the void fraction inside the reactor core decreased. The decline of the void fraction will increase the reactivity feedback; hence, the power increased rapidly until the reactor scram. Further, with safety relief valves open, the dome pressure would not exceed the criteria regulated by ASME. In addition, to cover the insufficiency of thermal hydraulic code, fuel rod transient analysis code FRAPTRAN was applied. To perform fuel rod transient analysis, thermal information from TRACE code would be entered into FRAPTRAN with power history. In addition, DAKOTA code was applied to concern the influence of fuel rod manufacturing tolerance. With uncertainty bands from DAKOTA analysis, the criteria could be judged with more confidence. This research successfully established a procedure of thermal hydraulic and fuel rod property analysis. Further, with SNAP interface, the FRAPTRAN and DAKOTA were combined successfully to perform the uncertainty analysis.

FOREWORD

The 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. 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 usually used in the nuclear power plants analysis. This report showed TRACE can also do the calculation of small system such as dry-storage cask.

Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program) which includes the development and maintenance of TRACE. INER (Institute of Nuclear Energy Research, Atomic Energy Council, R.O.C.) is the organization in Taiwan responsible for the application of TRACE in thermal hydraulic safety analysis, for recording user's experiences of it, and providing suggestions for its development. To meet this responsibility, this research established a procedure of thermal hydraulic, fuel rod property and uncertainty analysis by using TRACE/FRAPTRAN/DAKOTA/SNAP for Kuosheng Nuclear Power Plant.

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EXECUTIVE SUMMARY

According to the user manual, TRACE is the product of a long term effort to combine the capabilities of the U.S. NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool. U.S. NRC has ensured that TRACE will be the main code used in thermal hydraulic safety analysis in the future without further development of other thermal hydraulic codes, such as RELAP5 and TRAC. Besides, the 3-D geometry model of reactor vessel, which is one of the representative features of TRACE, can support a more accurate and detailed safety analysis of NPPs. TRACE also can provide greater simulation capability than the previous codes, especially for events like LOCA.

FRAPTRAN (Fuel Rod Analysis Program Transient) 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). In this research, FRAPTRAN version 1.4 was applied. With FRAPTRAN 1.4, the important criteria such as cladding temperature, cladding hoop strain, fuel enthalpy and radial gap can be determined.

DAKOTA (Design Analysis Kit for Optimization and Terascale Applications) is a tool kit which is developed by Sandia National Laboratory (SNL). It supplies a simple setting interface for researchers to perform different iteration method with the analysis code. From the DAKOTA manual, the uncertainty quantification can be determined by several simulations with different input parameter sets. That is, with the comparison and computation of the input parameter distribution and output data variation, the uncertainty can be determined.

SNAP (Symbolic Nuclear Analysis Package) 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 text files, the graphical control blocks and thermal hydraulic connections make researches comprehend the whole power plant and control system more easily. Further, without any patch files, the SNAP interface allow researchers connect the DAKOTA uncertainty parameters with the analysis codes directly. It is a powerful tool to do uncertainty calculations more efficiently.

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 will start SPU from Cycle 24 and Unit 2 will start SPU from Cycle 23. The operating power will be 3030 MWt. In order to uprate the power, the assessments for the fuel rods mechanical properties under the SPU conditions are required. This research established a procedure of thermal hydraulic, fuel rod property and uncertainty analysis by using TRACE/FRAPTRAN/DAKOTA/SNAP. The overpressurization hypothetical accidents were analyzed in this study.

From these analysis data, the Kuosheng NPP will be safe during the overpressurization transient after SPU. In spite that turbine bypass valve failed, the dome pressure would not exceed the criteria regulated by ASME if certain amounts of safety/relief valves opened. Further, the closure time of valves in this research were shorter than that in reality. The hypothetical transient conditions were more severe. However, the reactor scram setting and other safety mechanism could still maintain the integrity of the NPP and the fuel rods. On the other hand, for the uncertainty analysis, the manufacturing tolerance might cause 1.7% of cladding temperature deviation, 14% of cladding hoop strain deviation and 10% of fuel enthalpy deviation. Nevertheless, even concerning the uncertainty deviation, the fuel rod properties would not exceed the criteria.

ABBREVIATIONS

ASME	American Society of Mechanical Engineers
BWR	Boiling-Water Reactor
cal/g	calorie(s) per gram
DAKOTA	Design Analysis Kit for Optimization and Terascale Applications
FRAPTRAN	Fuel Rod Analysis Program Transient
GWd/MTU	Gigawatt-days per metric ton of uranium
g	gram(s)
INER	Institute of Nuclear Energy Research Atomic Energy Council, R.O.C.
K	Kelvin
kg	kilogram(s)
kW	kilowatt(s)
LOCA	Loss of Coolant Accidents
MPa	Megapascal(s)
MSIVC	Main Steam Isolation Valves Closure
MUR	Measurement Uncertainty Recapture
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
Pa	Pascal(s)
PCT	Peak Cladding Temperature
PCMI	Pellet Cladding Mechanical Interaction
PDFs	Probability Distributions Functions
PNNL	Pacific Northwest National Laboratory
SNAP	Symbolic Nuclear Analysis Program
SNL	Sandia National Laboratory
SPU	Stretch Power Uprated
SRV	Safety/Relief Valves
TCVC	Turbine Control Valves Closure
TRACE	TRAC/RELAP Advanced Computational Engine
TSVC	Turbine Stop Valves Closure
W	Watt(s)
US	United States
μm	micrometer(s)

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 will start SPU from Cycle 24 and Unit 2 will start SPU from Cycle 23. The operating power will be 3030 MWt. In order to uprate the power, the assessments for the fuel rods mechanical properties under the SPU conditions are required. This research established a procedure of thermal hydraulic, fuel rod property and uncertainty analysis by using TRACE/FRAPTRAN/DAKOTA/SNAP. The overpressurization hypothetical accidents were analyzed in this study.

TRACE (TRAC/RELAP Advanced Computational Engine) is a thermal hydraulic analysis code which is developed by Apt Inc. and conducted by U.S. NRC. It can perform the Loss of Coolant Accidents (LOCAs), operational transients or other accident scenario with best-estimate analyses [3]. Different from other traditional thermal hydraulic code, in the TRACE code, the component Vessel which can automatically considered the three dimensional heat transfer is applied to simulate reactor vessel. With this component, researchers can easily assign how many sections, radial rings and height level are needed.

FRAPTRAN (Fuel Rod Analysis Program Transient) 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) [4]. In this research, FRAPTRAN version 1.4 was applied. With FRAPTRAN 1.4, 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 [5]. 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.

DAKOTA (Design Analysis Kit for Optimization and Terascale Applications) is a tool kit which is developed by Sandia National Laboratory (SNL). It supplies a simple setting interface for researchers to perform different iteration method with the analysis code. From the DAKOTA manual, the uncertainty quantification can be determined by several simulations with different input parameter sets. That is, with the comparison and computation of the input parameter distribution and output data variation, the uncertainty can be determined [6].

SNAP (Symbolic Nuclear Analysis Package) 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 text files, the graphical control blocks and thermal hydraulic connections make researches comprehend the whole power plant and control system more easily. Further, without any patch files, the SNAP interface allow researchers connect the DAKOTA uncertainty parameters with the analysis codes directly. It is a powerful tool to do uncertainty calculations more efficiently.

2. METHODOLOGY

For the fuel rod transient analysis in FRAPTRAN code, the input parameters required can be basically divided into three parts including materials of rods, geometry of rods and thermal hydraulic conditions. The previous two parts can be obtained from the FSAR and Taiwan Power Company training documents. The thermal hydraulic conditions, however, are hard to be determined especially in hypothetical accidents. Typically, conditions of some similar plant transients would be used. Nevertheless, it is hard to find the transient conditions quite the same as the hypothetical accident. To overcome this difficulty, in this research, the thermal hydraulic code TRACE is performed for gathering thermal conditions, which would be closer to the hypothetical accidents [7]. Then, the thermal conditions would be input into the FRAPTRAN code to do the further fuel rods analysis. In addition, the manufacturing tolerance of fuel rods had been concerned. Together with the DAKOTA uncertainty job stream in SNAP interface, the output data results of FRAPTRAN code would not be a single curve but an uncertainty band which varied with time. Due to the DAOKOTA uncertainty iterations, there would be several FRAPTRAN computations performed and generate a large amounts of data results. As a result, a data extraction program, which can extract a specific interesting parameters from all data results in a short time, was developed by our group [8]. To conclude, the flowchart of this methodology is shown in Figure 1.

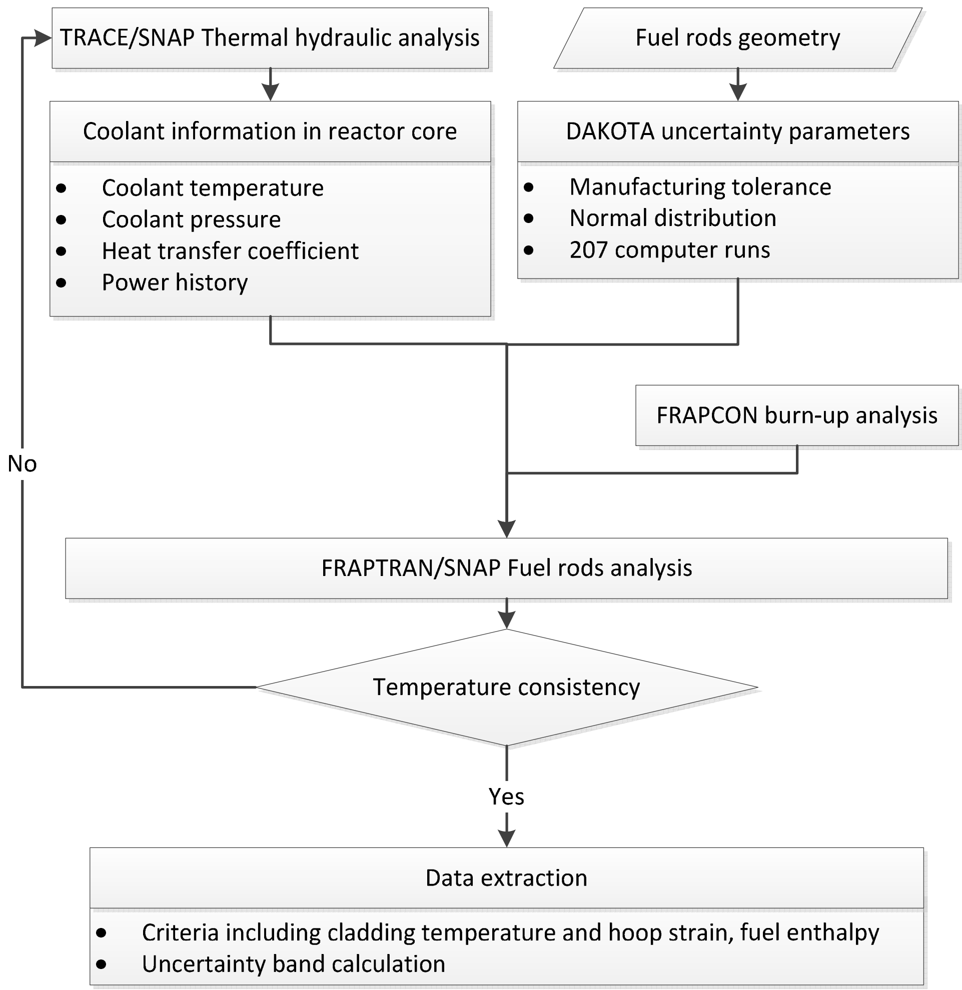


Figure 1 Flow Chart of the Uncertainty Analysis

3. MODEL ESTABLISHMENT

3.1 TRACE Thermal Hydraulic Model

TRACE model of Kuosheng NPP was built up according to the FSAR, design documents and TRACE manual. Further, only the NSSS was considered in this TRACE model. As a result, the turbine was set as boundary conditions in this research. The power plants system outside the turbine would not be considered in this model, as shown in Figure 2. From the comparison of TRACE steady state data results and the operating conditions as shown in Table 1, it is noticed that the TRACE model suitable for the Kuosheng NPP. Details of the pressure vessel, reactor core, feedwater controlling system, water level computation and power simulation of this analysis model are described as follows:

Vessel

The pressure vessel is simulated by the 3-D component Vessel which includes down comers, fuel assemblies, upper and lower plenum. This component are divided into 2 sections, 4 radial rings and 11 height levels. Details of the vessel component is shown in Figure 3.

Channel

The fuel assemblies of Kuosheng NPP are simulated by six 1-D components "Channel", which were applied for simulating the flow inside and besides the fuel assemblies. Each Channels was divided into 27 axial nodes, 6 radial nodes. There are 96 to 120 fuel assemblies were set up in these six channels respectively. Further, inside the fuel assemblies, there are three kinds of fuel rods including full-length, half-length and water rods. Details of the rods arrangement and properties were set up according to the FSAR and shown in Figure 4.

Feedwater Control System

Three feedback signals including water level, steam flow rate and feedwater flow rate are set up in this TRACE model. As these three signals deviate the normal operating conditions, the controlling system will adjust the valves open of the feedwater. Moreover, although the water level can be measured directly through the control blocks in the TRACE model, the NRWL is also built up referred to the reality in the TRACE model, as shown in Figure 5.

Power

In the TRACE code, there are three neutron feedback modes including Power Table, Point kinetics or combined with PARCS. The point kinetics mode with reactivity feedback table is chosen for Kuosheng NPP in this TRACE model. In addition to the reactivity feedback, decay heat should also be concerned. There are also several decay heat groups that have been built in. In this model, the ANS-73 was chosen. Details of the power components setting are shown in Figure 6 and Figure 7.

Steam Lines

According to the configuration of Kuosheng NPP, there are four main steam pipe lines in the TRACE model. On the main steam pipe lines, there are main steam line isolation valves, turbine stop valves and turbine control valves. Further, 16 safety/relief valves are also set up on the main steam pipe lines. At the ends of the pipe lines, the Break 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 built in this model as shown in Figure 2. 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 8 and Figure 9.

Pumps

Basically, there are three kinds of pumps including feedwater pump, recirculation pump and jet pump to maintain the circulation of the reactor core. In this model, due the assumption of the boundaries, two feedwater pumps in this model are simulated by the components Fills. These Fills are connected to valves which flow area could be changed to control the feedwater flow rate. These valves then would be connected to the vessel component at the position radial ring 4, section 1 and level 7.

There are 2 recirculation pumps shown in Figure 2. In front of the outlet of each recirculation pump, there is a control valve which can be used for controlling the recirculation flow rate. Further, a jet pump is connected on each recirculation loops to inject recirculation flow inside the reactor vessel at position radial ring 4, section 1 and level 6. Details of these pumps are shown in Figure 10 to Figure 12.

Table 1 Comparison of the Initial Conditions of FSAR and TRACE

Parameters	Unit	FSAR	TRACE model
Power	(MWt)	3030	3030
Dome Pressure	(MPa)	7.17	7.17
Feedwater Flow	(kg/sec)	1645.3	1641
Steam Flow	(kg/sec)	1645.3	1641
Core inlet flow	(kg/sec)	10645	10704.8
Recirculation flow (Single loop)	(kg/sec)	1549.5	1538.6
Core exit pressure	(MPa)	7.23	7.3
NWRL	(m)	0.934	0.934

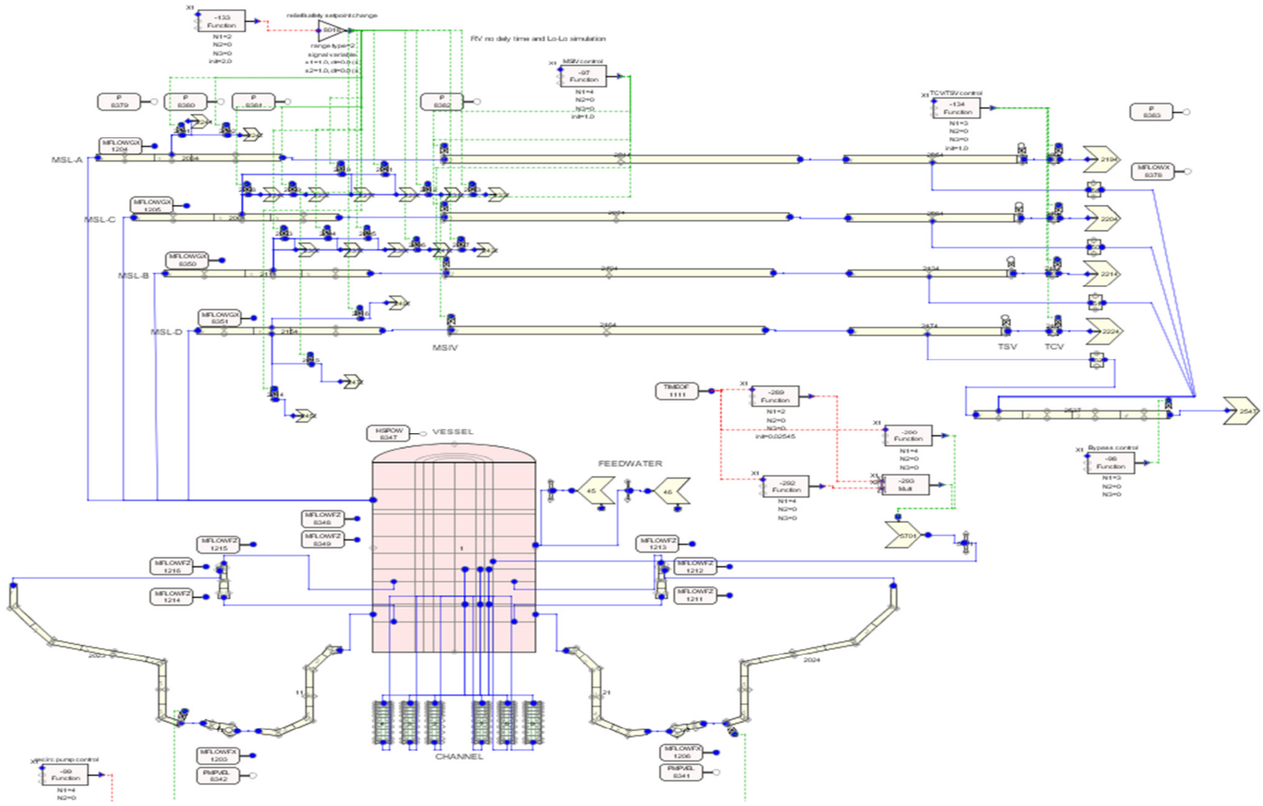


Figure 2 TRACE Model of Kuosheng NPP

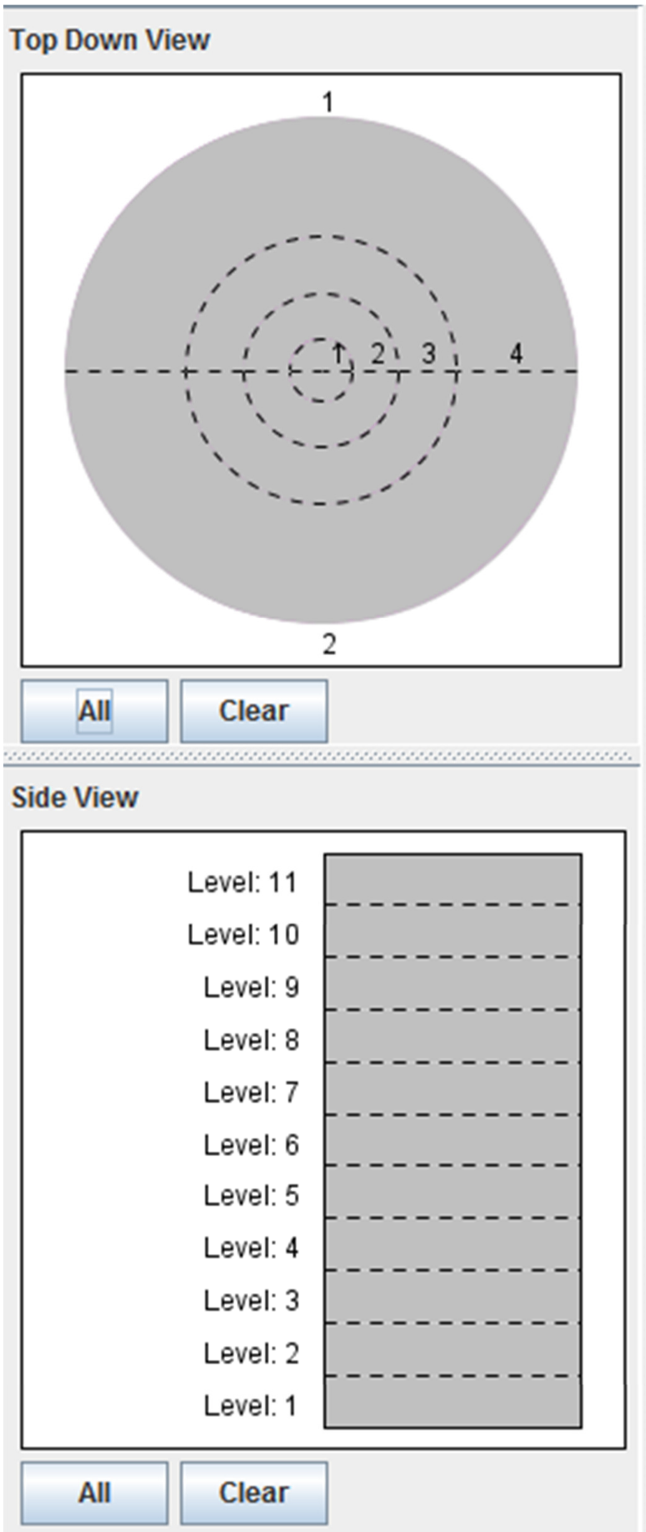


Figure 3 Vessel Geometry of Kuosheng NPP in the TRACE Model

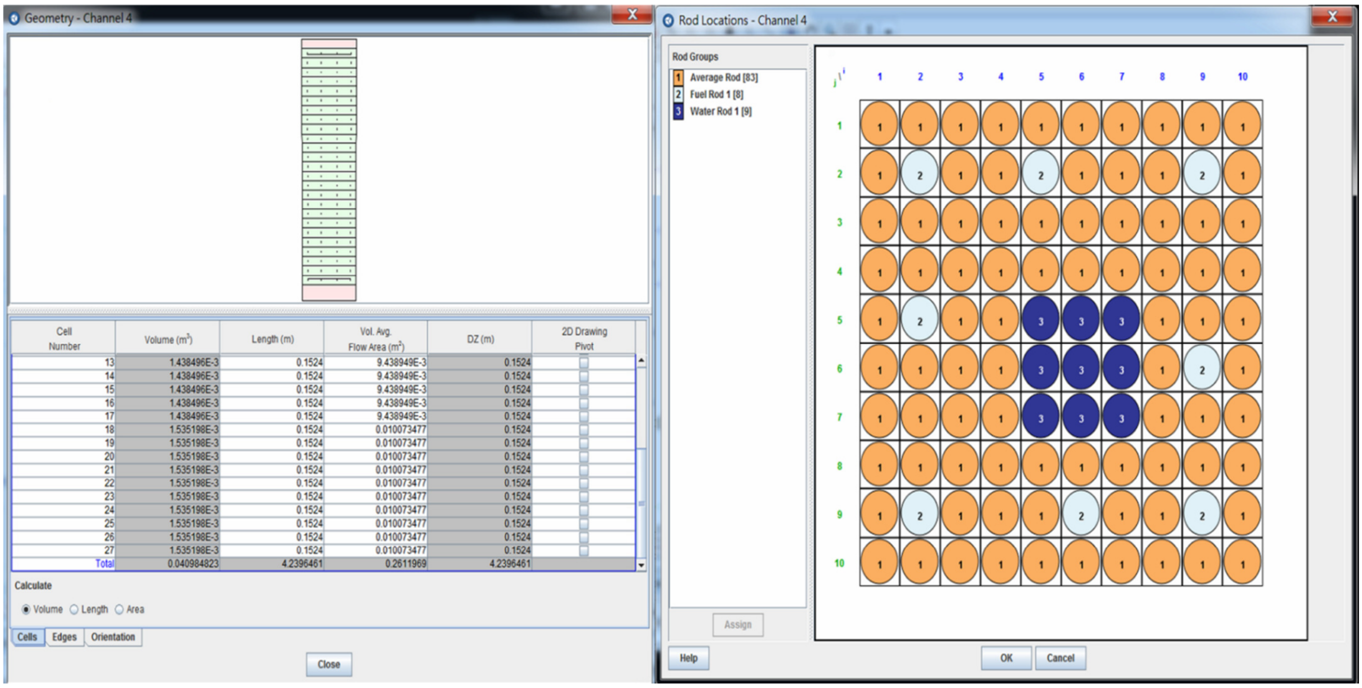


Figure 4 Fuel Rods Arrangement of Kuosheng NPP in TRACE Model

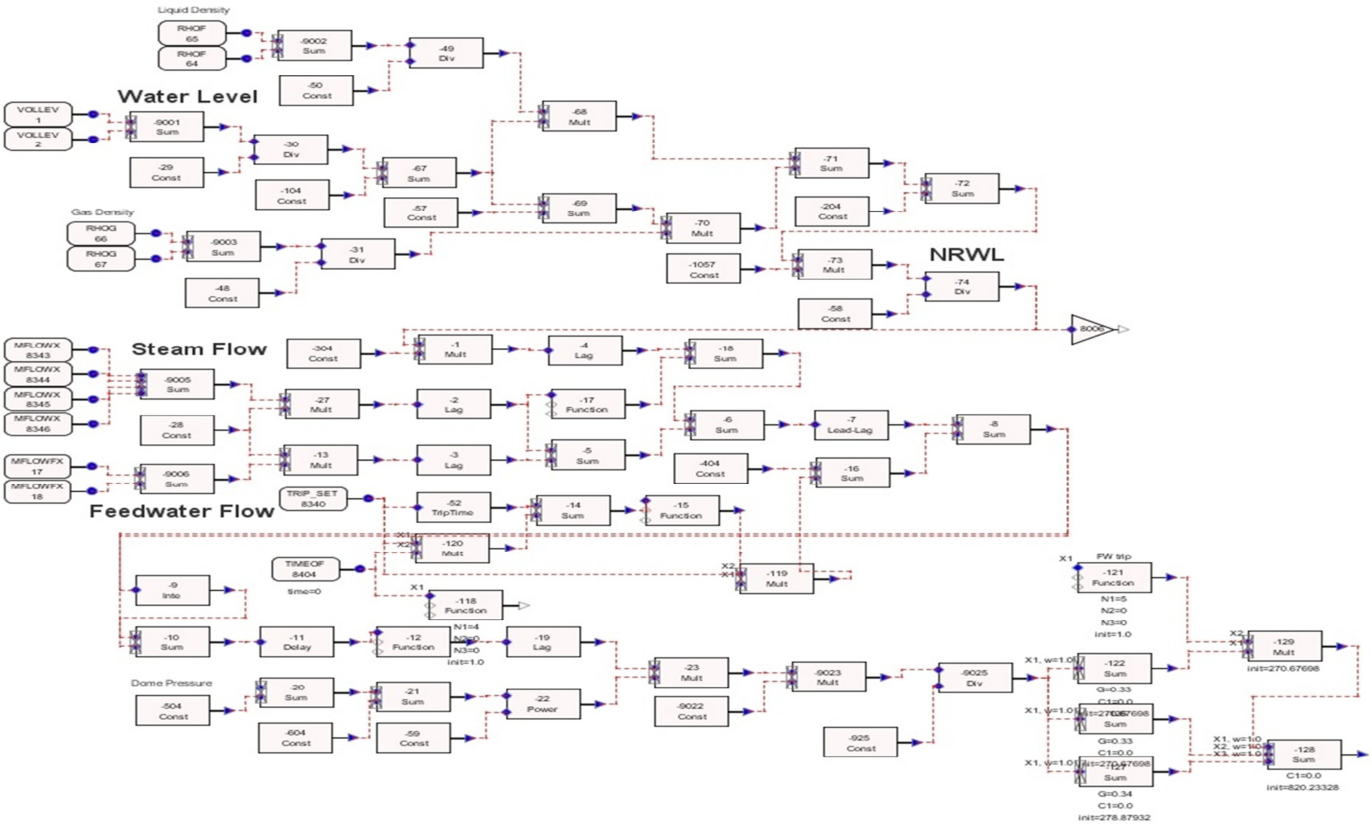


Figure 5 Feedwater Control System of Kuosheng NPP in TRACE Model

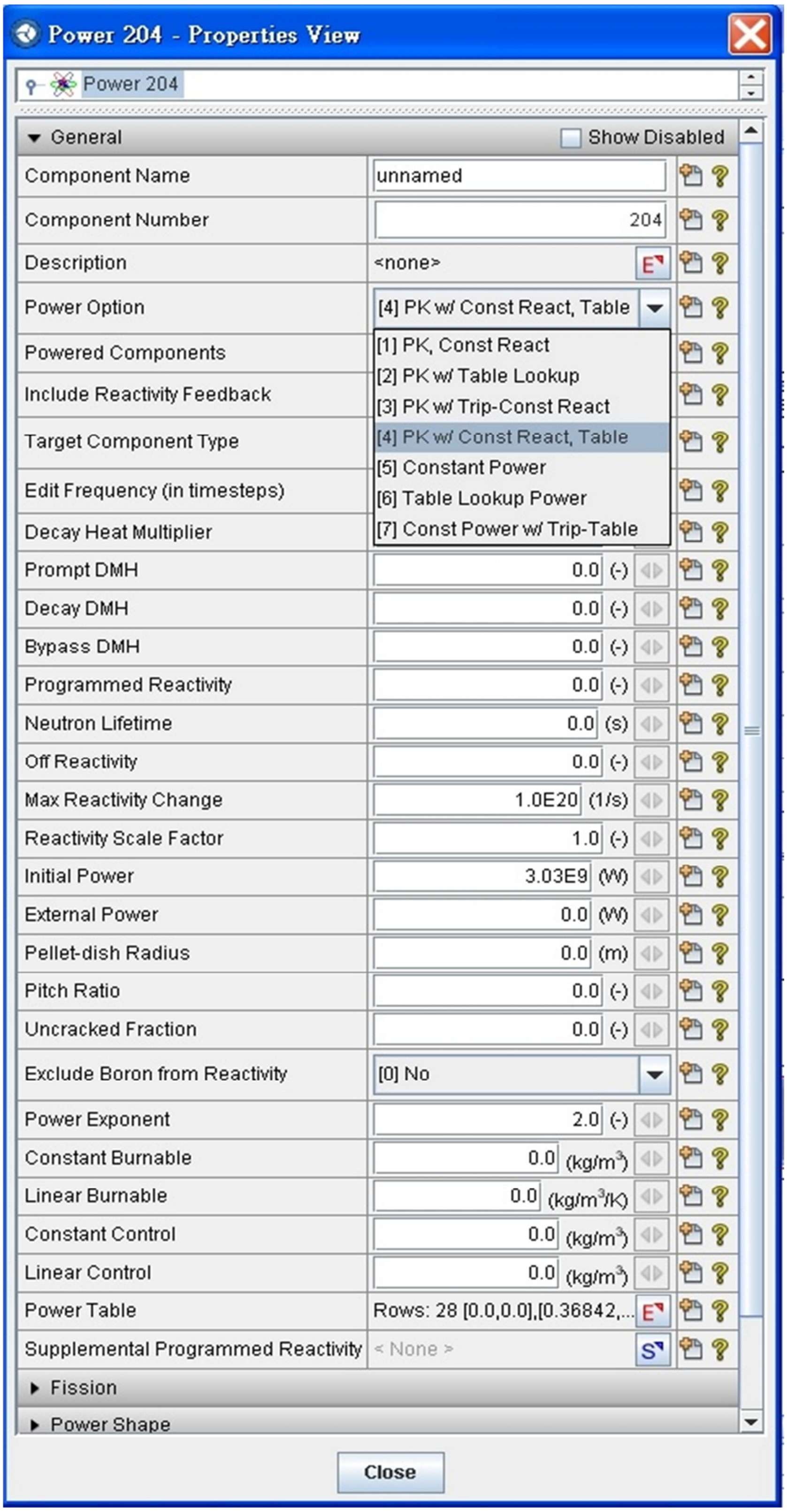


Figure 6 General Setting of Power Component of Kuosheng NPP in TRACE Model

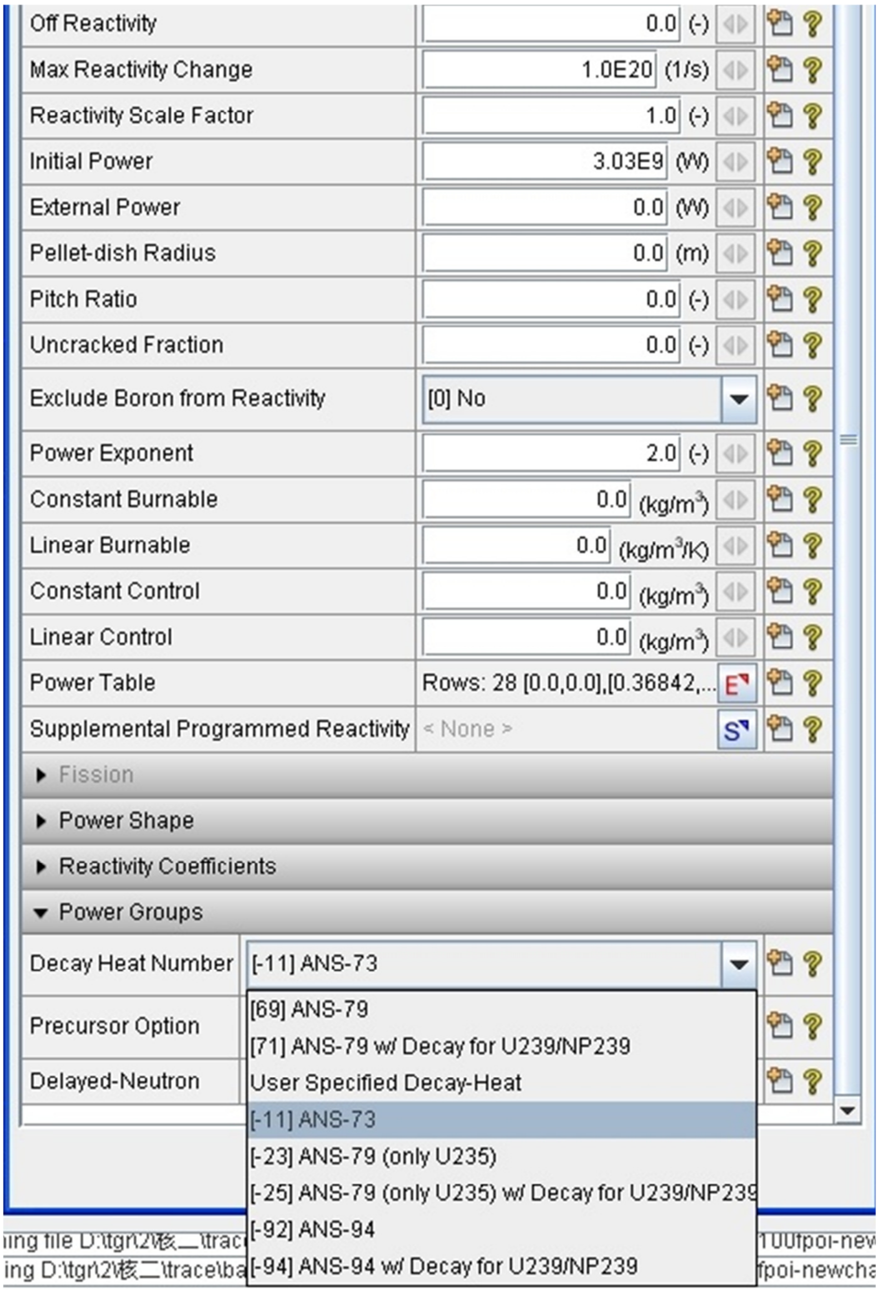
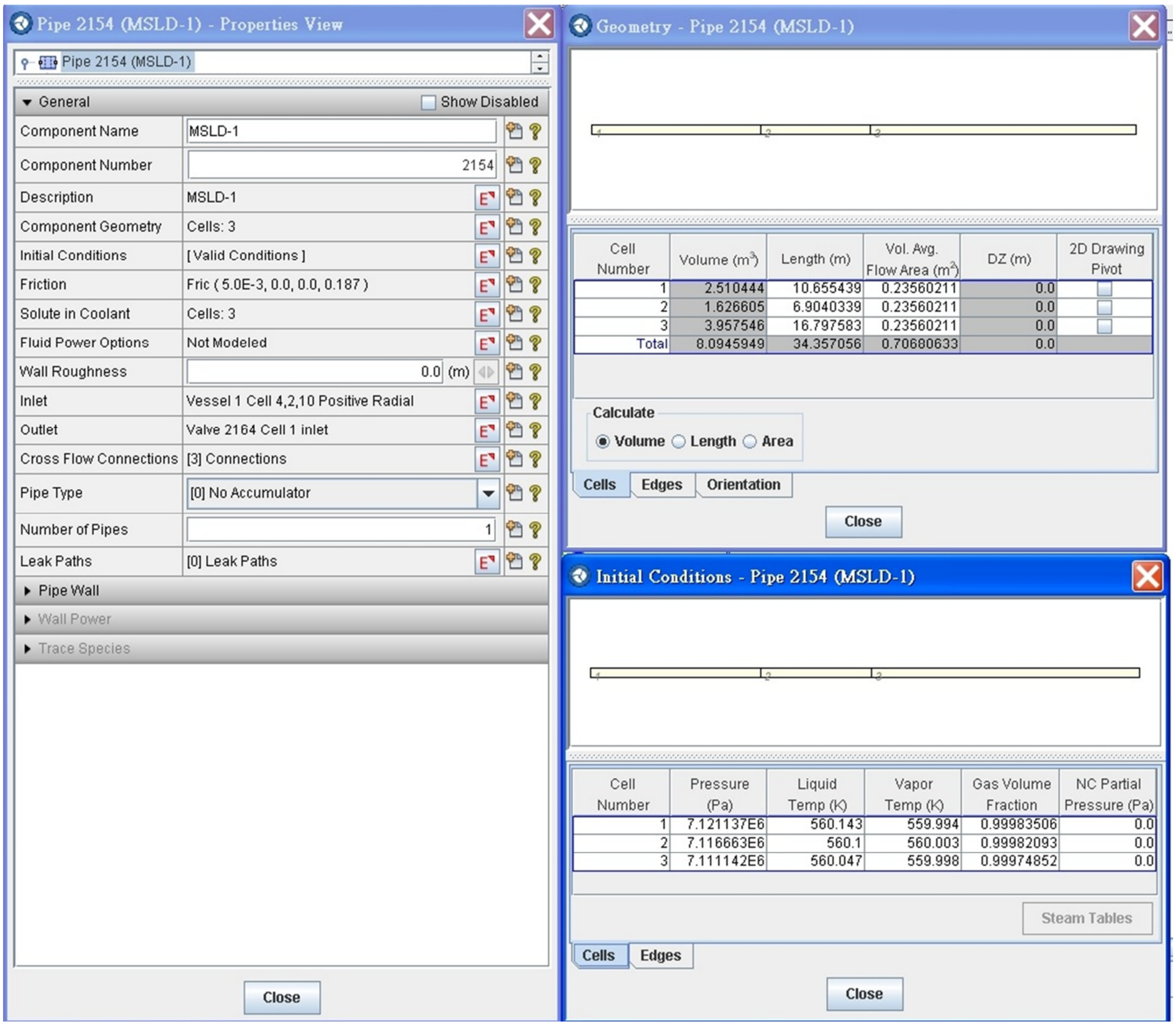


Figure 7 Decay Heat Setting of Kuosheng NPP in TRACE Model



Cell Number	Volume (m³)	Length (m)	Vol. Avg. Flow Area (m²)	DZ (m)	2D Drawing Pivot
1	2.510444	10.655439	0.23560211	0.0	
2	1.626605	6.9040339	0.23560211	0.0	
3	3.957546	16.797583	0.23560211	0.0	
Total	8.0945949	34.357056	0.70680633	0.0	

Cell Number	Pressure (Pa)	Liquid Temp (K)	Vapor Temp (K)	Gas Volume Fraction	NC Partial Pressure (Pa)
1	7.121137E6	560.143	559.994	0.99983506	0.0
2	7.116663E6	560.1	560.003	0.99982093	0.0
3	7.111142E6	560.047	559.998	0.99974852	0.0

Figure 8 Main Steam Pipe Lines Properties of Kuosheng NPP in TRACE Model

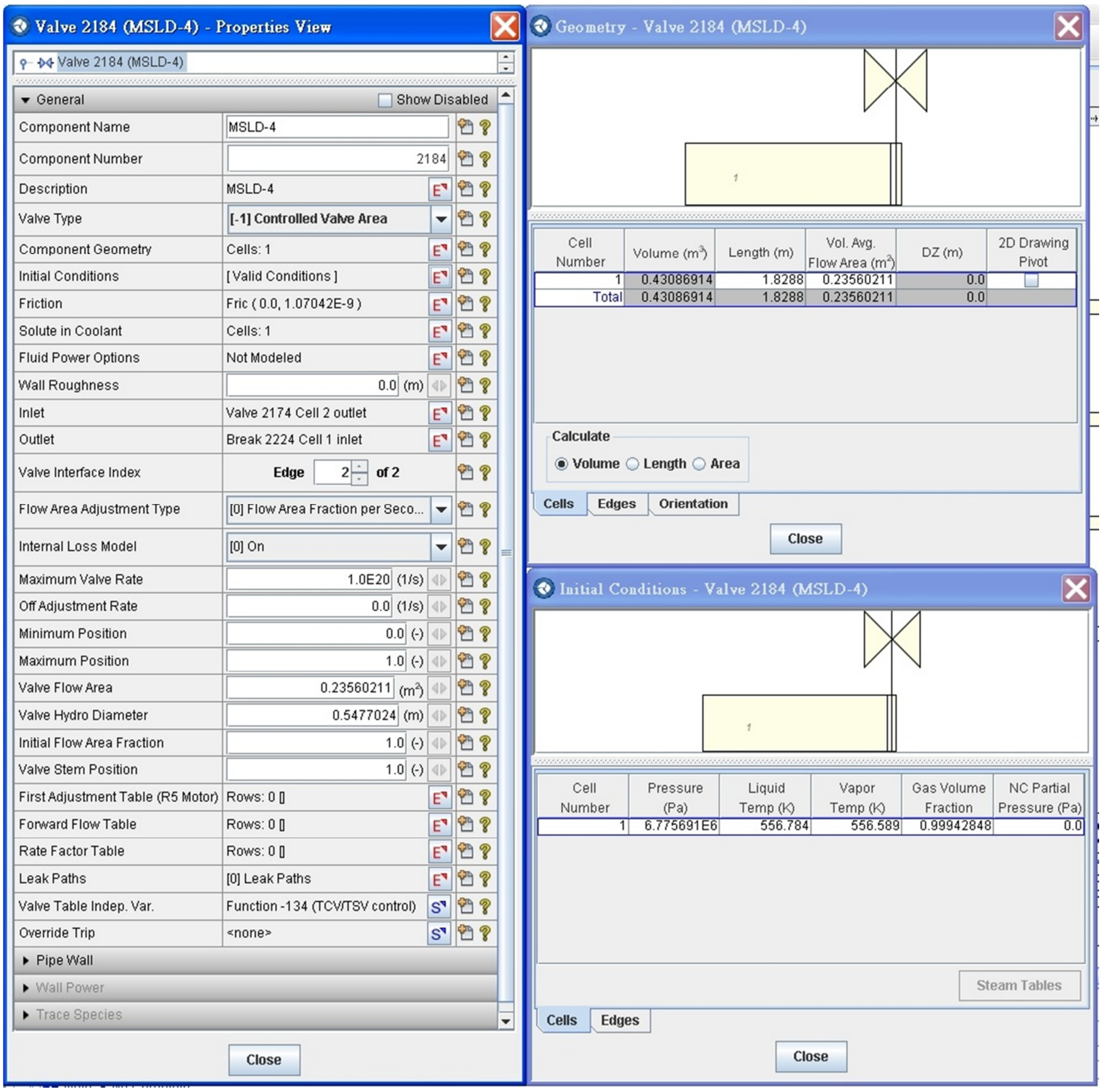


Figure 9 Properties of Main Steam Line Isolation Valve of Kuosheng NPP

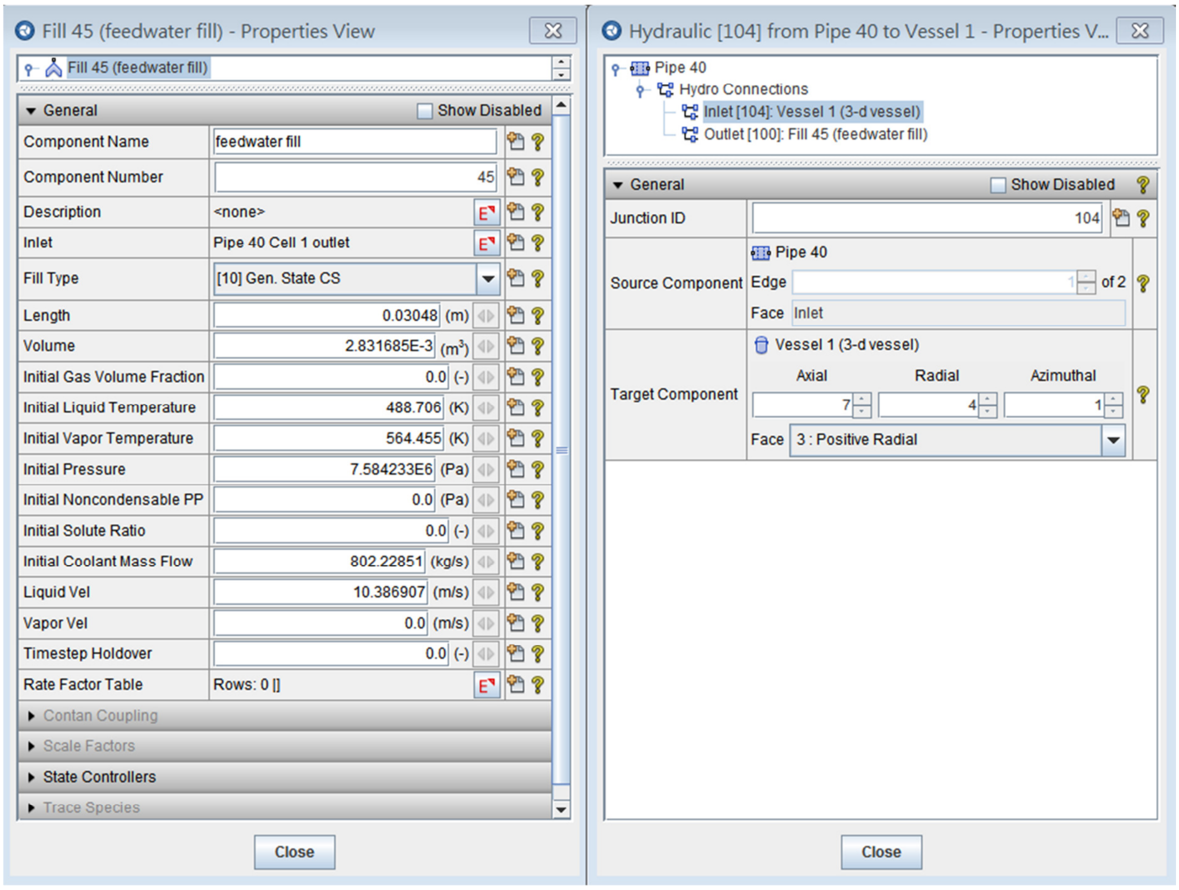


Figure 10 Feedwater Pumps Settings of the Kuosheng NPP in TRACE Model

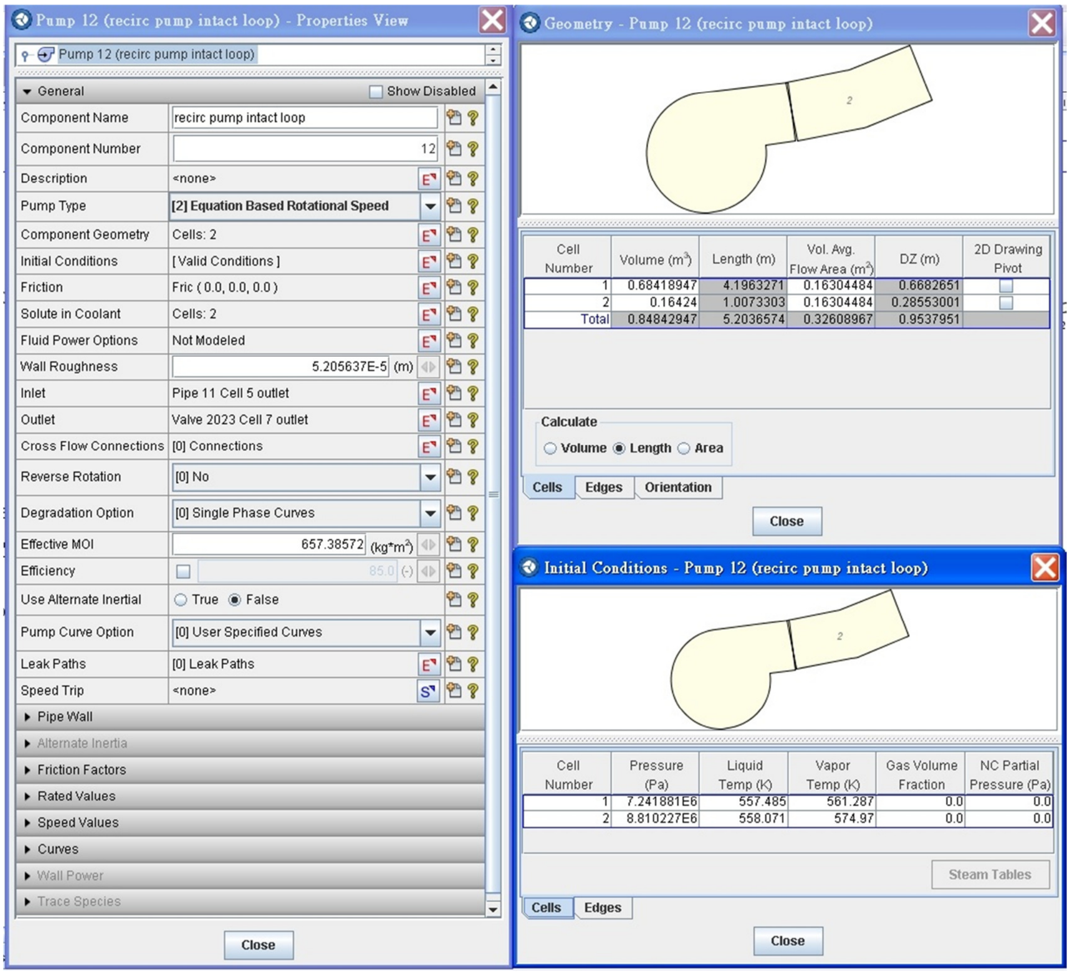


Figure 11 Recirculation Pump Properties of Kuosheng NPP in TRACE Model

3.2 FRAPTRAN Fuel Rod Model

Fuel Rod Geometry and Convergence Settings

The fuel rod geometric information was shown in Table 2. 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.

Fuel Rod Node

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 12 and Figure 13, 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 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 2 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	

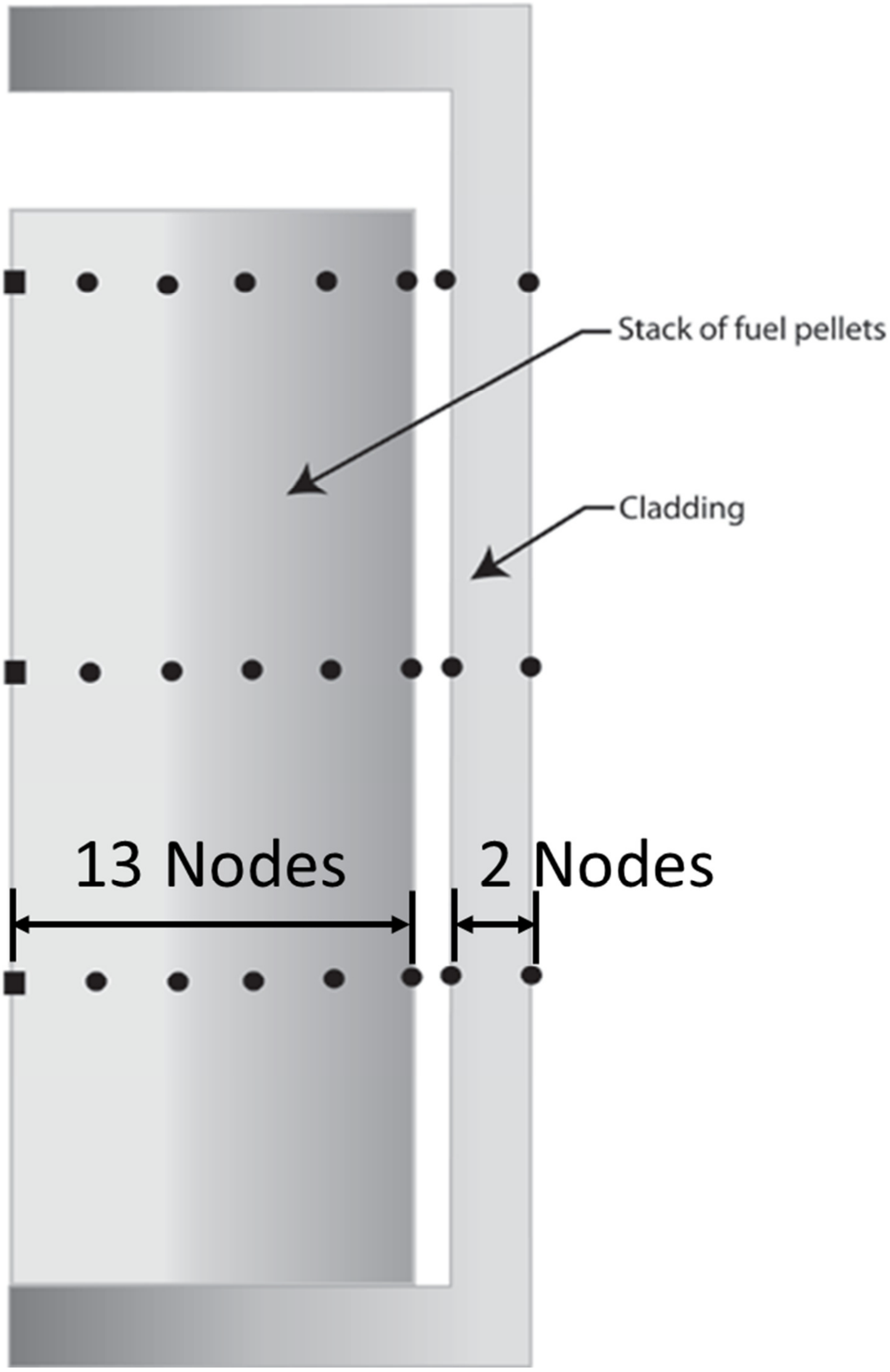


Figure 12 Setting of Radial Nodes in FRAPTRAN Model

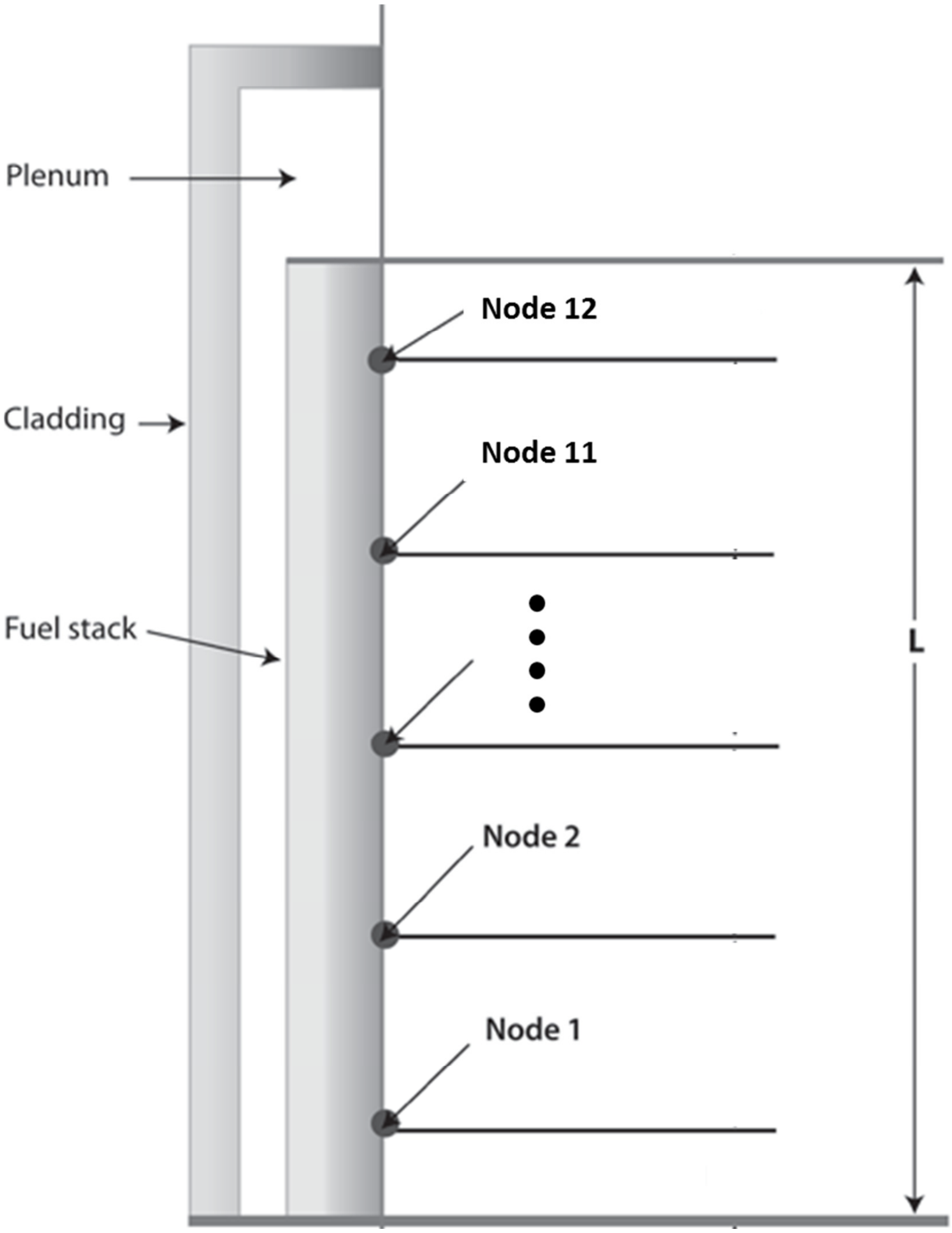


Figure 13 Setting of Axial Nodes in FRAPTRAN Model

3.3 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. As shown in Fig.14, the water level of reactor vessel and the main steam pipe lines, which color depth would be changed to illustrate variation of void fraction during transients, were connected to the TRACE data results. Inside the reactor vessel, there are three bars divided into 12 blocks, which connected to FRAPTRAN data results, to display different kind of fuel rod properties including cladding temperature, cladding hoop strain and fuel enthalpy. On the both side of this figure, there are 5 bars which can show the specific value of important criteria. Further, in the middle of this figure, there are initial conditions of Kuosheng NPP and events sequence of the specific transient. From the animation model, interactions among different parts of the NPP could be observed more easily. Moreover, combining data results of TRACE and FRAPTRAN codes is also a new concepts for NPP safety analysis.

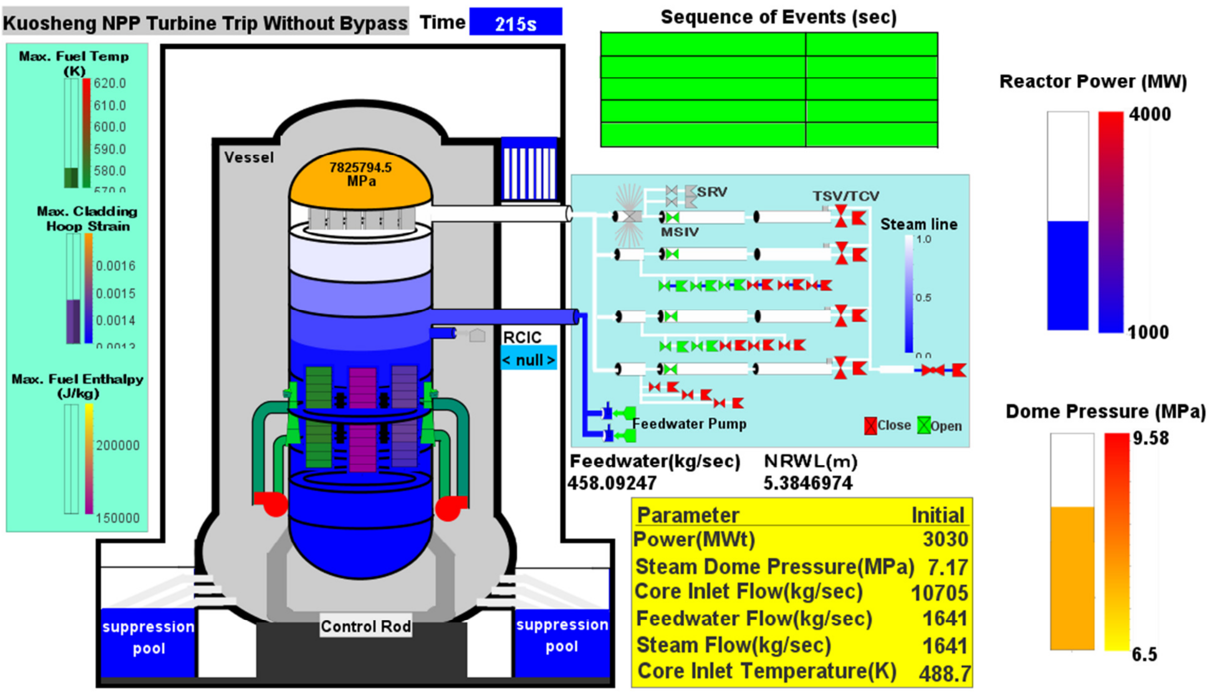


Figure 14 Animation Model of Kuosheng NPP

3.4 DAKOTA Uncertainty Analysis Model

In order to simplified analysis and reduce computational time, the influence of sub-model was not concerned in this research. Nonetheless, for conservative reason, all the sub-model such as Ballooning, BJK model were turned on. According to NUREG/CR-7001 report [9], which was prepared in 2009 by the FRPATRNA developing group, 6 manufacturing tolerances including cladding diameter, fuel pellet diameter, fuel pellet density, gap fill pressure, cladding roughness and fuel pellet surface roughness were concerned in this research. Table 3 lists the setting details of these uncertainty parameters. This table was set through the SNAP interface which is enable to connect DAKOTA variables with the FRAPTRAN analysis.

Table 3 Uncertainty Parameters Setting

Uncertainty parameters	Mean	Std.	Distribution
Cladding outer diameter (mm)	11.22	0.04	Normal
Cladding roughness (um)	1.00	0.30	Normal
Fuel pellet outer diameter (mm)	0.98	0.01	Normal
Fuel pellet density (% TD)*	94.50	0.91	Normal
Fuel pellet roughness (um)	1.00	0.3	Normal
Rod fill pressure (Pa)	515038.37	68999.97	Normal

After built up the FRAPTRAN model with the SNAP interface, the variables which could be connected with the DAKOTA code should be set up. On the left column of SNAP interface, as shown in Figure 15, there is a Numerics list which could define several types of variables such as integers, reals, Booleans, strings and so on. According to the parameters shown in Table 3, there are six variables, “real”, being defined in this research. After setting up the variables, as shown in Figure 16, there will be connection buttons next to the input box of the rod geometry and fuel pellet setting columns. After connecting the variables respectively, the ranges of variables should also be defined. As shown in Figure17, the Job stream types should be set as “DAKOTA Uncertainty” mode. Then, clicking the parameter properties button in the “Job stream” column as shown in Figure 18, the uncertainty parameters could be defined respectively according to Table 3.

In this research, all the uncertainty parameters were calculated at the same time. This method would be closer to the reality because the interaction of each parameters would be concerned. However, a large amount of data results would be generated; as a result, the well-established statistical concepts and data extraction method should be applied. In order to process the data files, a data extraction program was developed by our group.

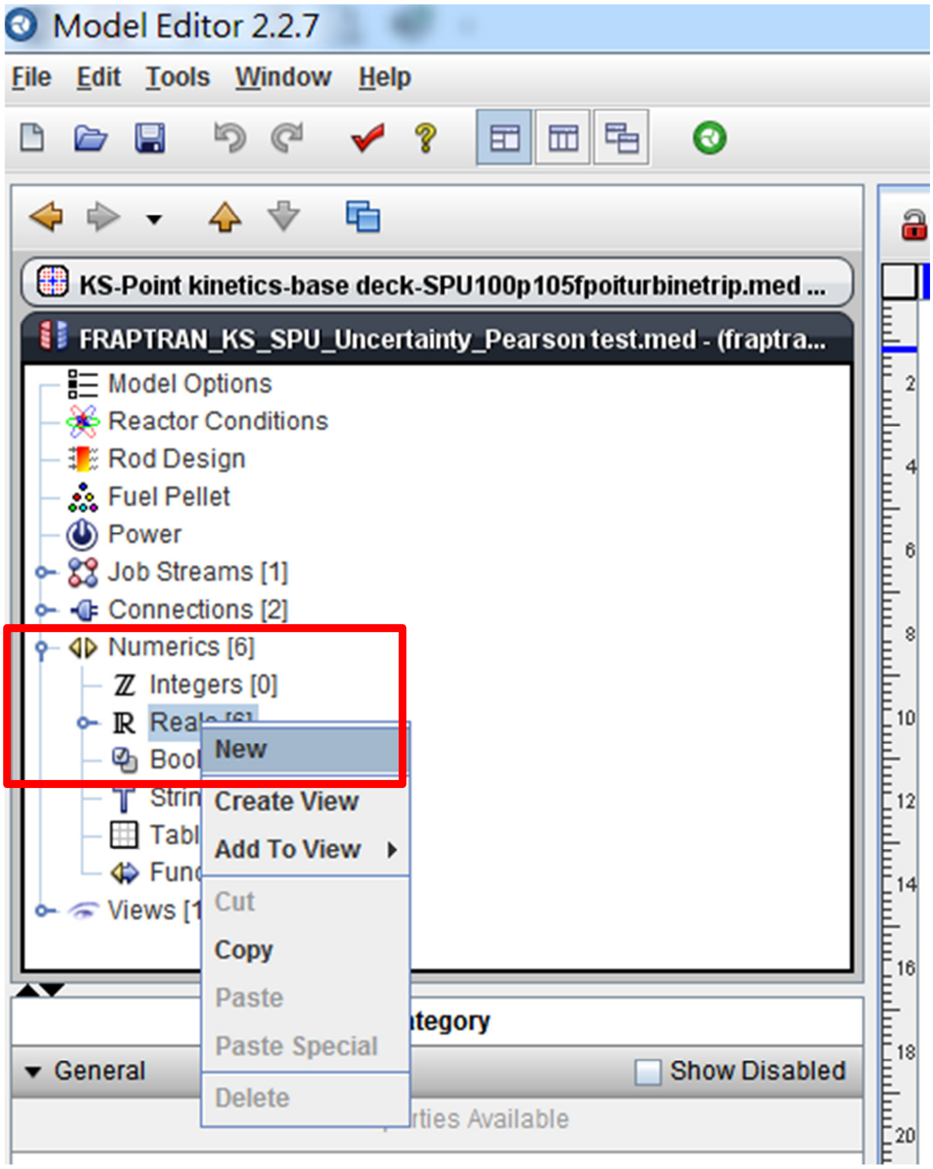


Figure 15　Numeric Variable Setting of DAKOTA Analysis in SNAP Interface

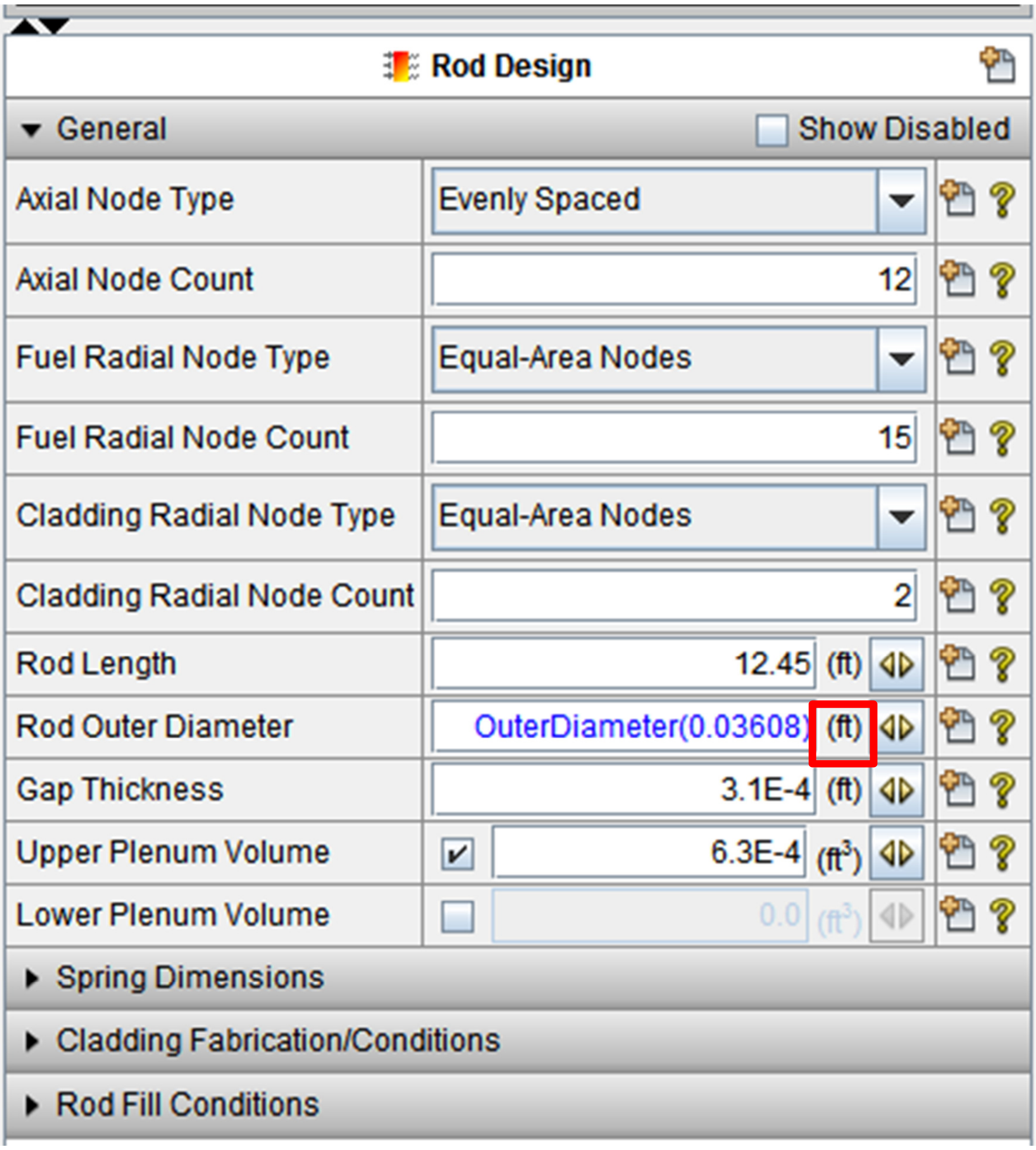


Figure 16 Connection of the Numeric Variables and the Rod Properties

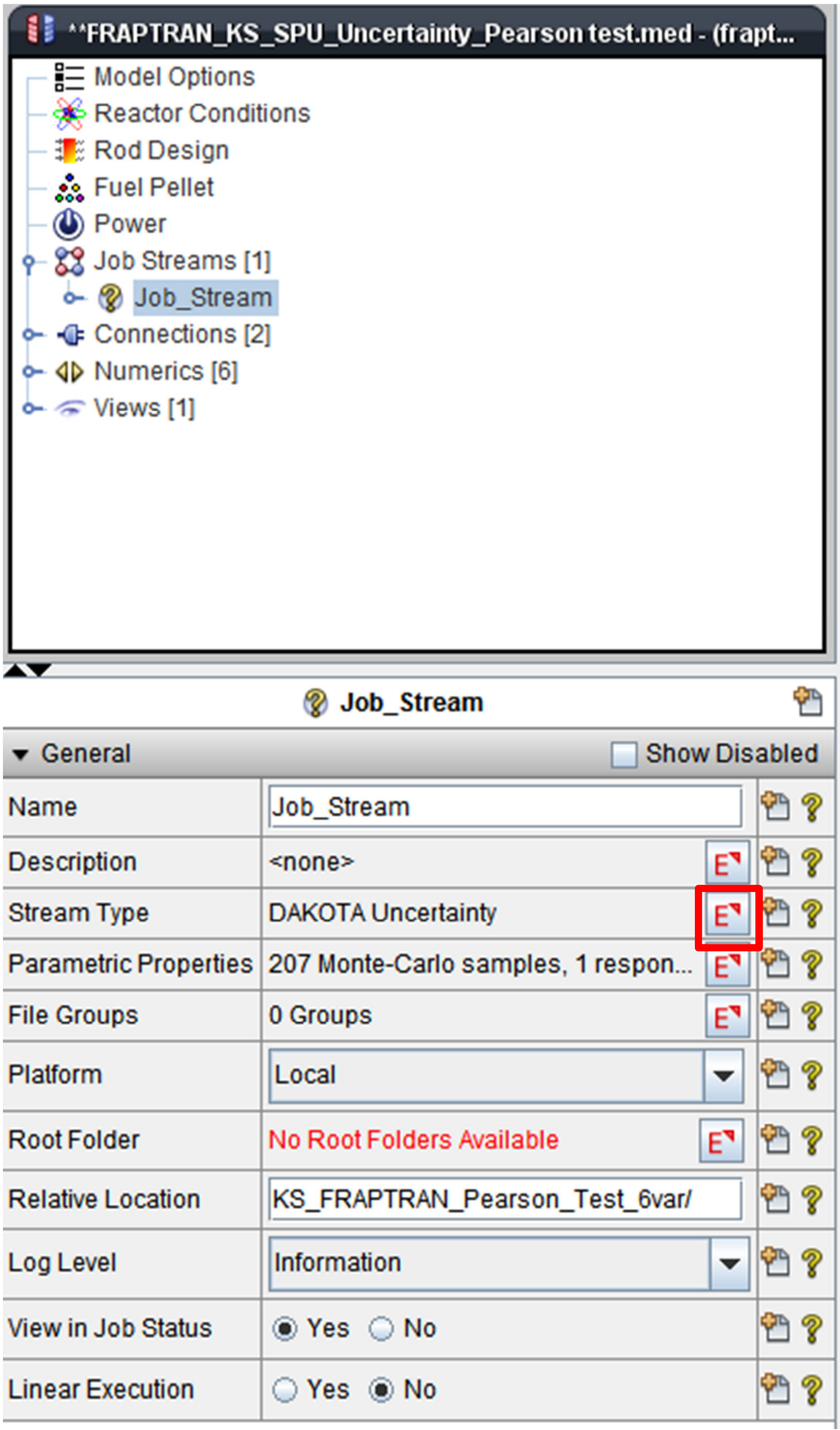


Figure 17 Uncertainty Job Stream Setting in SNAP Interface

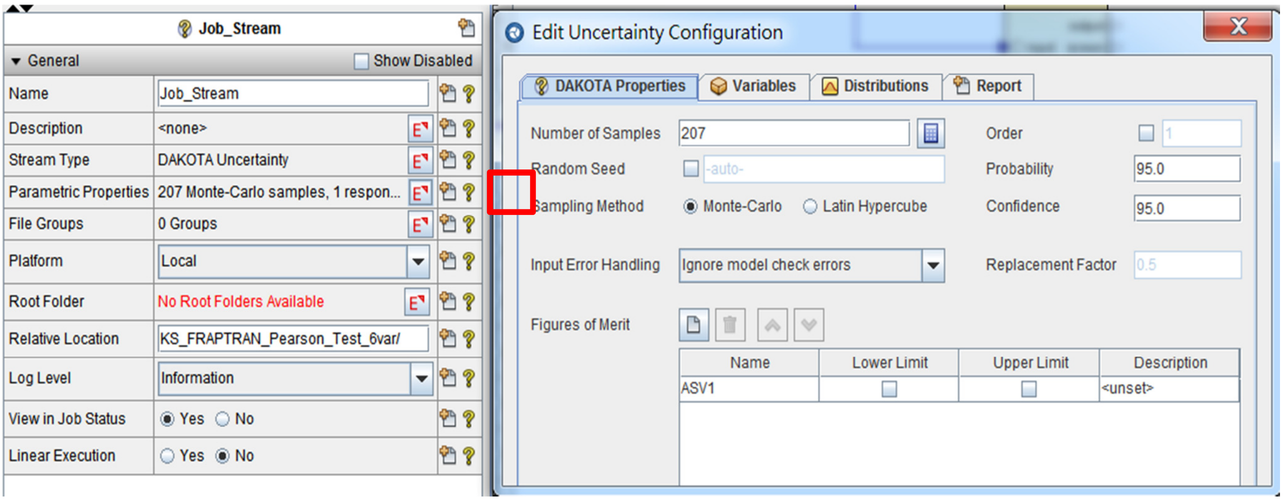


Figure 18 Variable Setting of DAKOTA Analysis in SNAP Interface

4. RESULTS AND DISCUSSION

According to the regulation of American Society of Mechanical Engineers (ASME), transients including main steam line isolation valve closure (MSIVC), turbine stop valve closure (TSVC) and turbine control valve closure (TCVC) have the potential to cause maximum peak system pressure. For conservative reason, turbine bypass valve in hypothetical accidents mentioned above would not open; further, setpoints of recirculation trip and safety/relief valves open would be higher than the real operating conditions in the NPP.

In the past works, the hypothetical scenario would be applied in the Kuosheng NPP TRACE model. Data results calculated by TRACE model could be compared with the criteria. According to the Boiler and Pressure Vessel Code by ASME, pressure of reactor vessel should not exceed 110% of the design value during transients, that is:

$$\text{Safety criteria of dome pressure} = 1.10 \times 8.62 \text{ MPa} = 9.58 \text{ MPa}$$

Moreover, because FRAPTRAN code was also applied, criteria of the fuel rods would be more important in this research. Due to the variation of the reactivity and increasing of the power during the transient, fuel rod criteria of Reactivity Induced Accident (RIA) were referred in this research. According to the NUREG 0800, during the RIA transient, the cladding temperature should not exceed 1088K; the cladding hoop strain should not exceed 0.01; the fuel enthalpy should not exceed 710600 J/kg (170 cal/g) and the oxidation layer increasing should not be larger 17% of the initial oxidation thickness. Further, the gap thickness between fuel pellets and fuel cladding should also be notice to prevent the Pellet Cladding Mechanical Interaction (PCMI). Based on these criteria and regulations, safety of the Kuosheng NPP during the overpressurization transients could be determined.

4.1 Main Steam Line Isolation Valves Closure (MSIVC)

Transient Assumption

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 to the plant system, the closure time in this case was shorter than 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. 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 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 4 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 4 Event Sequence of MSIVC Transient

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

Analysis Data Results

The MSIVs started to close at 210 second and fully closed at 213 second. As a result, Figure 19 shows that the steam flow rate greatly decreased 2 second later the transient started. Further, at 213 second, the MSIVs fully closed and the dome pressure increased greatly. As a result, the void fraction inside the reactor core decreased which increased the power greatly as shown in Figure 20. This figure also plotted two scram conditions including the 122% reactor power and dome pressure 7.66 MPa. From this figure, it is known that the reactor scrammed because of the excessive power.

Despite the fact that the reactor had scrammed at 213.23 second, the decay heat still heated the reactor vessel and generated steam. As a result, the dome pressure kept increasing. As the dome pressure reached to 7.82 MPa, the recirculation pumps tripped as shown in Figure 21. 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 as shown in Figure 22. Further, because the dome pressure varied near the setpoints, the safety valves opened and closed for three times.

From the TRACE data, 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 transient.

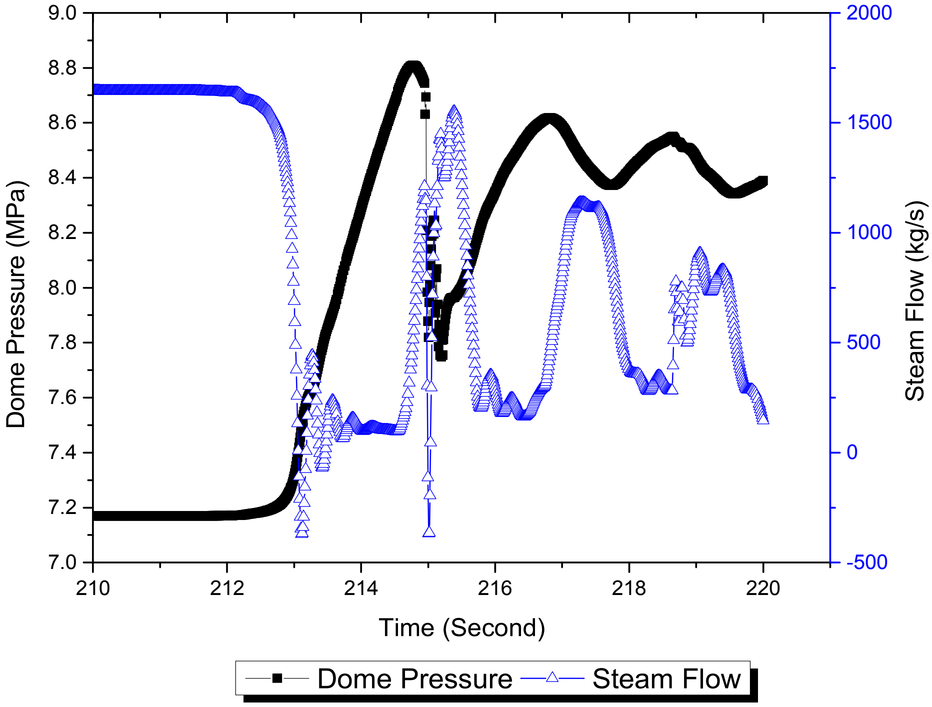


Figure 19 Variation of Dome Pressure and Steam Flow Rate in MSIVC Transient

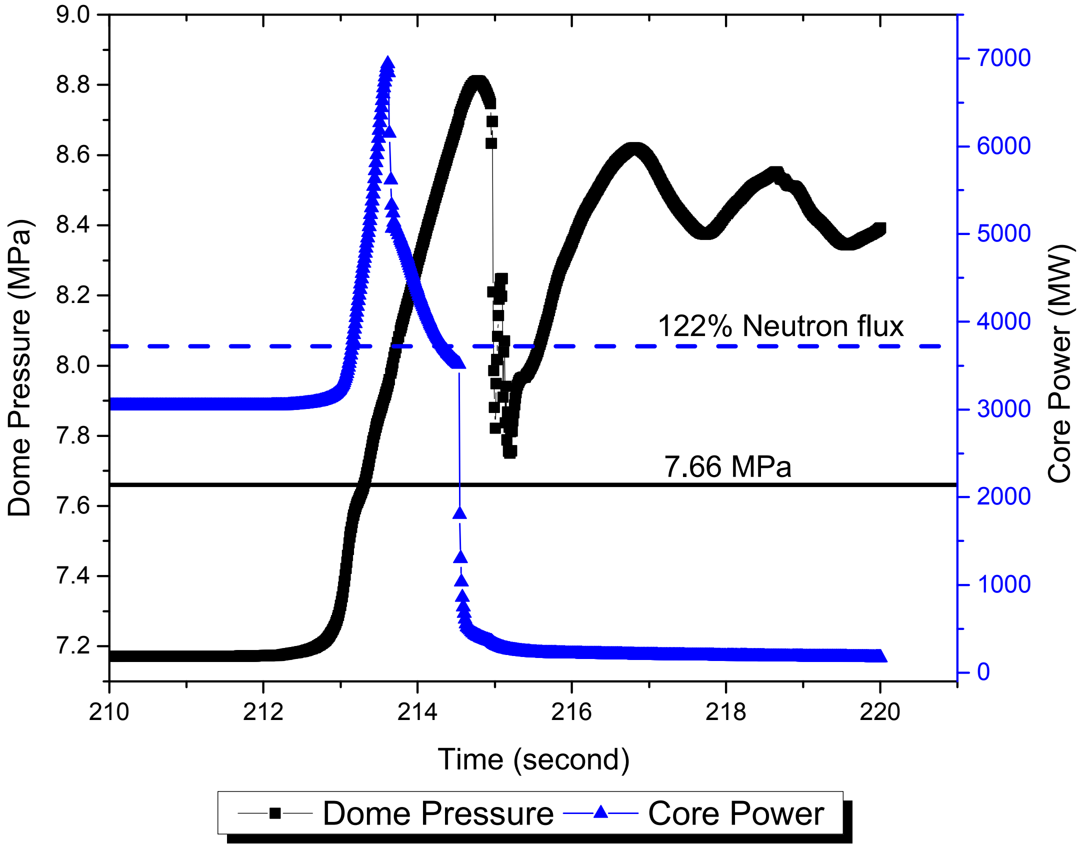


Figure 20 Variation of Dome Pressure and Core Power in MSIVC Transient

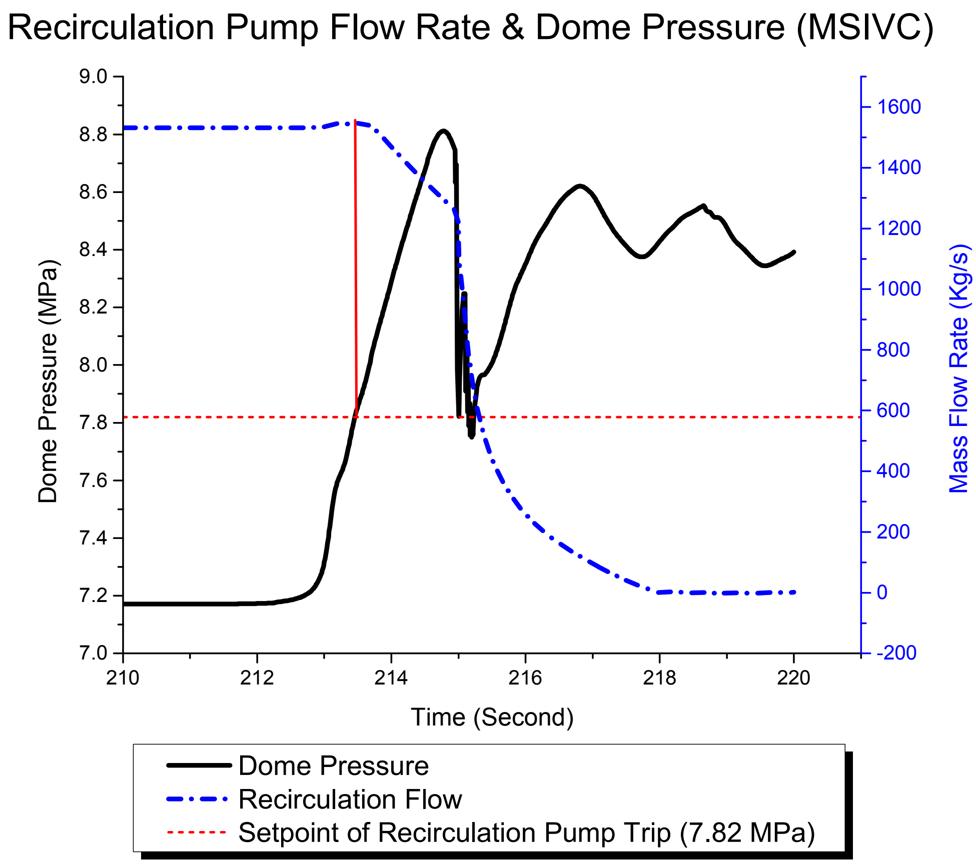


Figure 21 Relationship between Recirculation Flow Rate and Dome Pressure

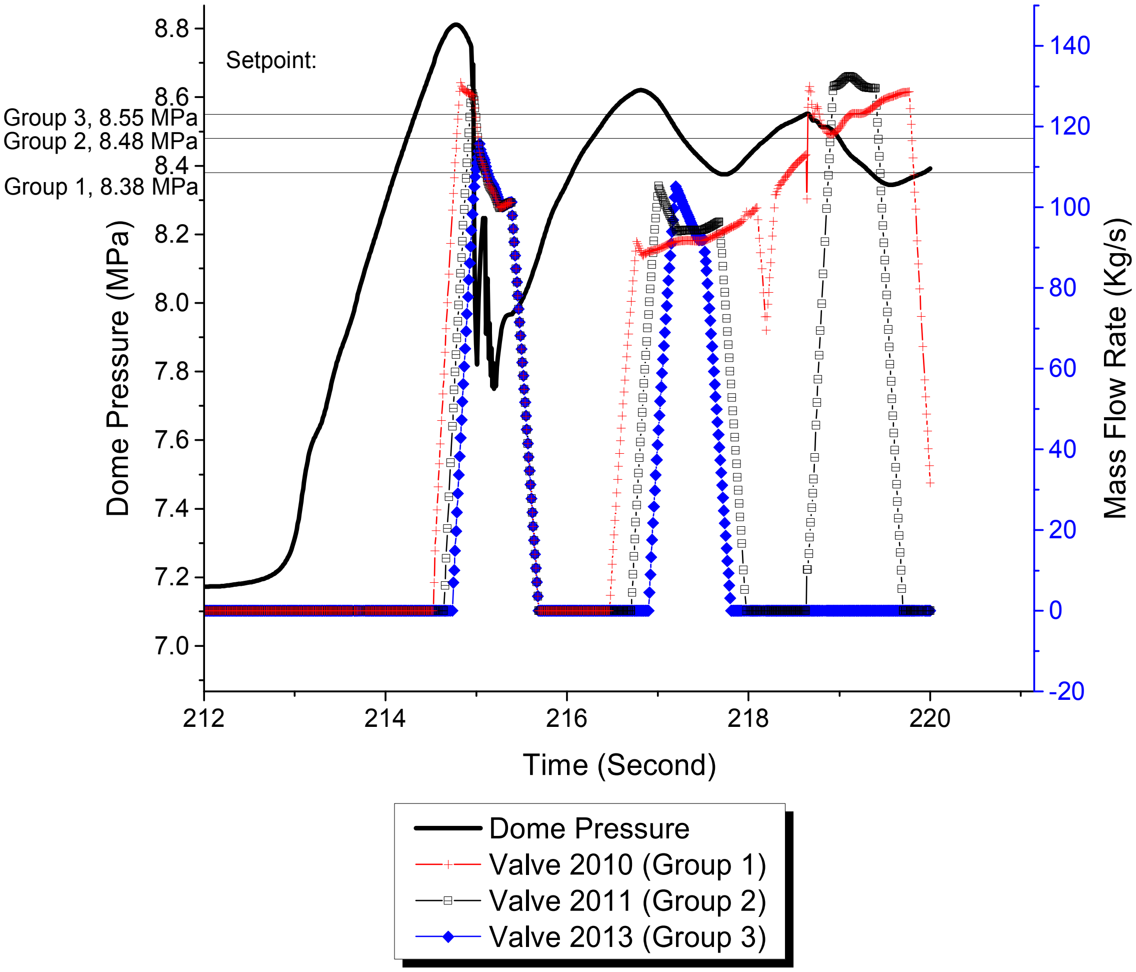


Figure 22 Relationship between Dome Pressure and Safety Valves Flow Rates

So far, only the thermal hydraulic properties were discussed. Details of the fuel rods such as burnup information and mechanical properties could not be calculated by the TRACE code. Figure 23 shows the average cladding temperature variation during the MSIVC transient. During the MSIVC transient, the cladding temperature increased because of the increasing power. However, the cladding temperature of Node 5 to Node 8 appeared higher value because of the power distribution of the fuel rods. Further, cladding temperature of Node 1 was much lower than that of other Nodes because of the good heat transfer conditions.

Figure 24 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. On the other hand, the increasing dome pressure would squeeze the fuel cladding, especially at the lower position such as Node 1. To conclude, the hoop strain vibrated during the transient because the thermal hoop strain and elastic hoop strain opposed to each other. From Figure 24, it is obvious that the cladding hoop strain would be at most 0.00154 which is much lower than the criteria because of the high coolant pressure.

In addition to the cladding hoop strain, the fuel pellet enthalpy should also be noticed in overpressurization transient. Figure 25 shows the enthalpy variation during the transient. The peak value is about 165 kJ/kg (39.47 cal/g), which is much lower than the criteria 170 cal/g. From those three criteria mentioned above, it is known that the fuel rods would keep good integrity during the MSIVC transient. However, some further parameters should be examined to ensure whether the water-metal reaction and the Pellet Cladding Mechanical Interaction (PCMI) happened or not. Figure 26 shows oxide thickness of the inside cladding. From this figure, it is noticed that there was no oxidation growth during the transient. On the other hand, from Figure 27, it is noticed that gap distance between cladding and fuel pellet was always larger than 0.064mm, which means that there is no PCMI.

In addition to the general FRAPTRAN analysis, uncertainty analysis of fuel rod properties are also concerned in this research. To make the uncertainty analysis results simple, there is only one uncertainty band with the highest value will be plotted in the figure. According to Figure23, the Node 5 was chosen as the representable in Figure 28. With the manufacturing tolerance uncertainties, the peak cladding temperature might increase up to 607K during the MSIVC transient. However, it is still much lower than the criteria 1088K even with the manufacturing tolerance concern. For the cladding hoop strain, as shown in Figure 29, the peak value of the uncertainty band is about 0.00154, which does not exceed the criteria 0.01. However, it should be noticed that the lower bound of the hoop strain uncertainty band is lower than zero, which means that the cladding might be squeezed. As a result, the gap distance between cladding and fuel pellet should be examined. According to the results in Figure 27, Figure 30 shows the uncertainty band of gap distance at Node 12. From this figure, it can be noticed that the gap distance is always larger than 0.52mm. The PCMI effect would not happen in MSIVC transient. Figure 31 shows the uncertainty band of fuel enthalpy at Node 7. With manufacturing uncertainties, the peak value is about 154000 J/kg, which is much lower than the criteria 170 cal/g. From these general and uncertainty analysis results of FRAPTRAN code mentioned above, the fuel rod still kept good integrity in MSIVC transient.

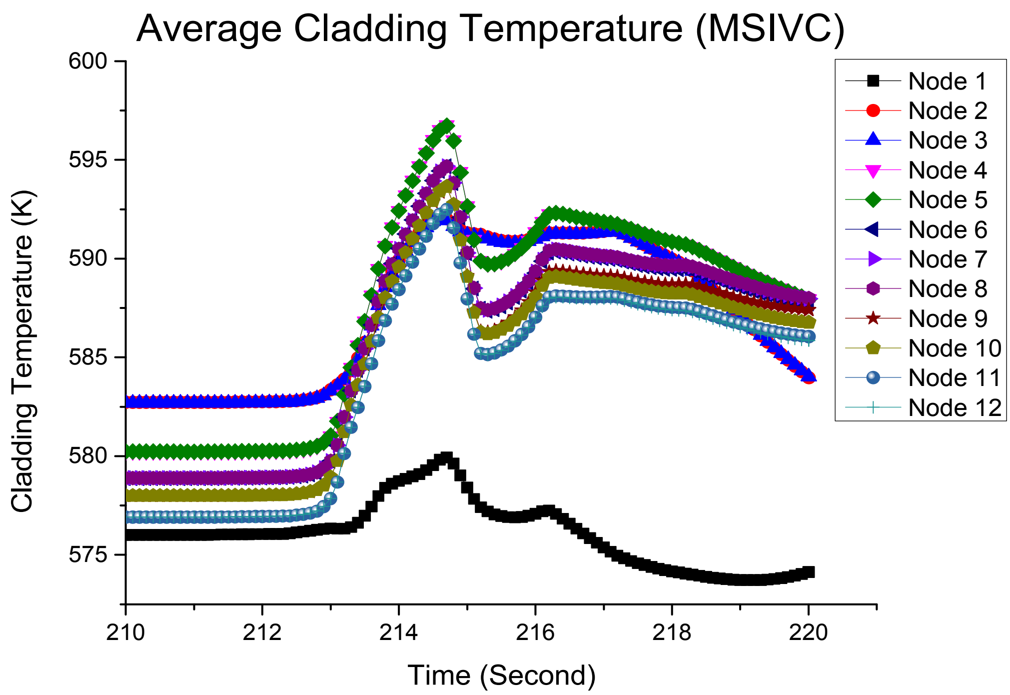


Figure 23 Cladding Temperature of FRAPTRAN Results in MSIVC Transient

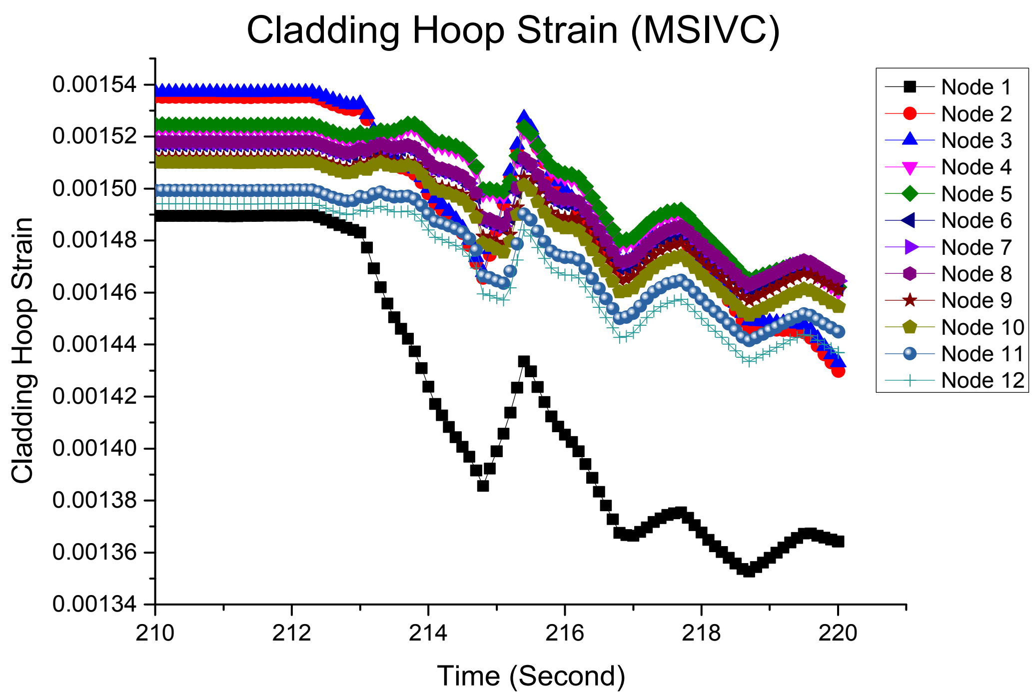


Figure 24 Cladding Hoop Strain of FRAPTRAN Results in MSIVC Transient

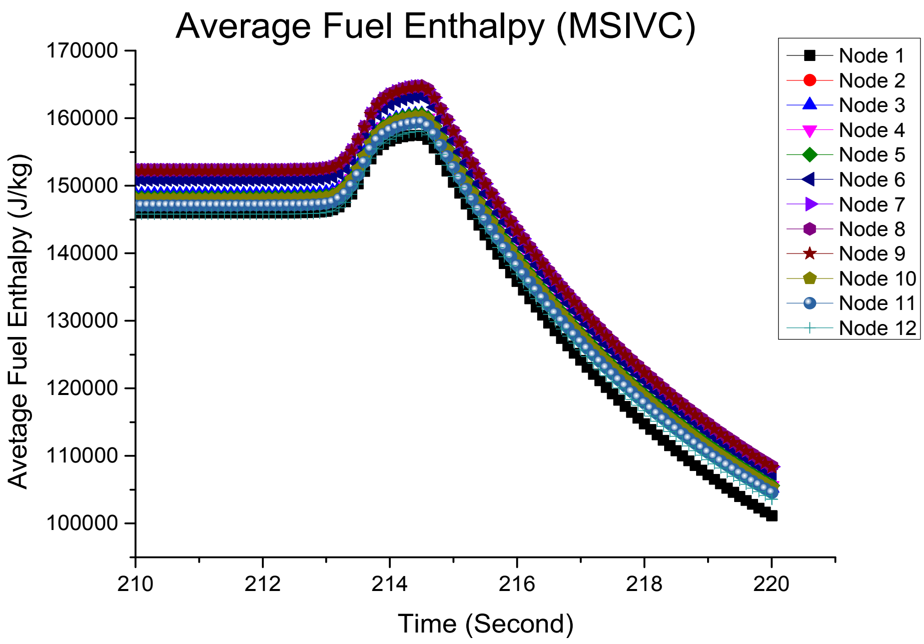


Figure 25 Fuel Enthalpy of FRAPTRAN Results in MSIVC Transient

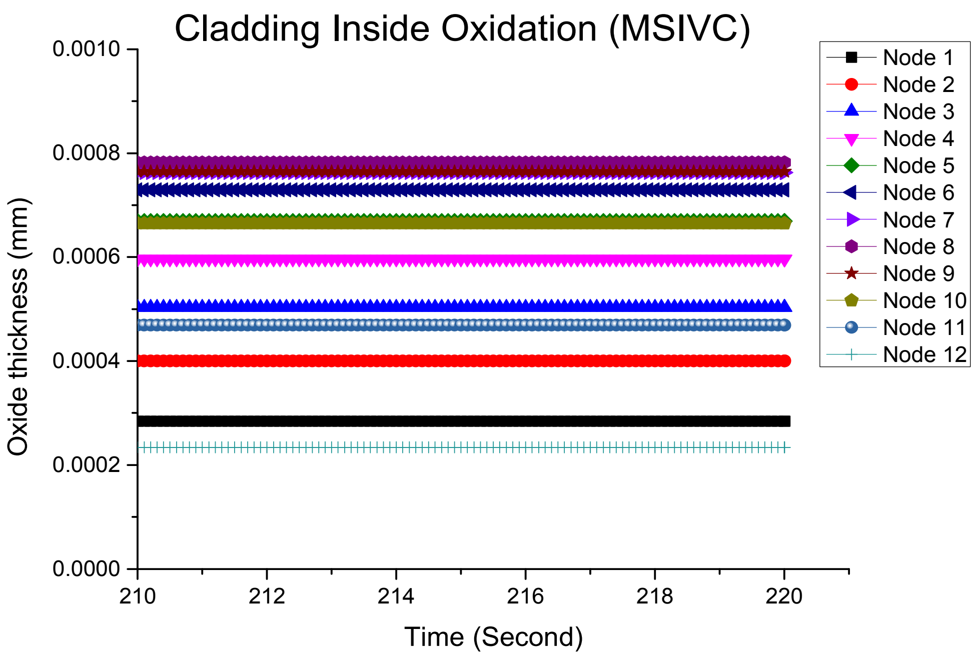


Figure 26 Oxidation Thickness of FRAPTRAN Results in MSIVC Transient

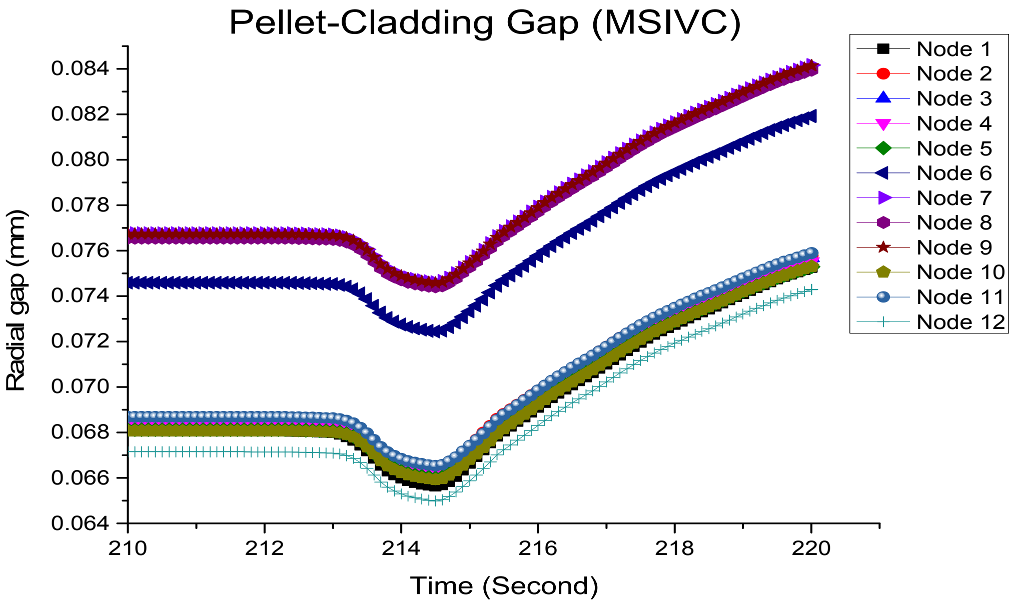


Figure 27 Gap Thickness of FRAPTRAN Results in MSIVC Transient

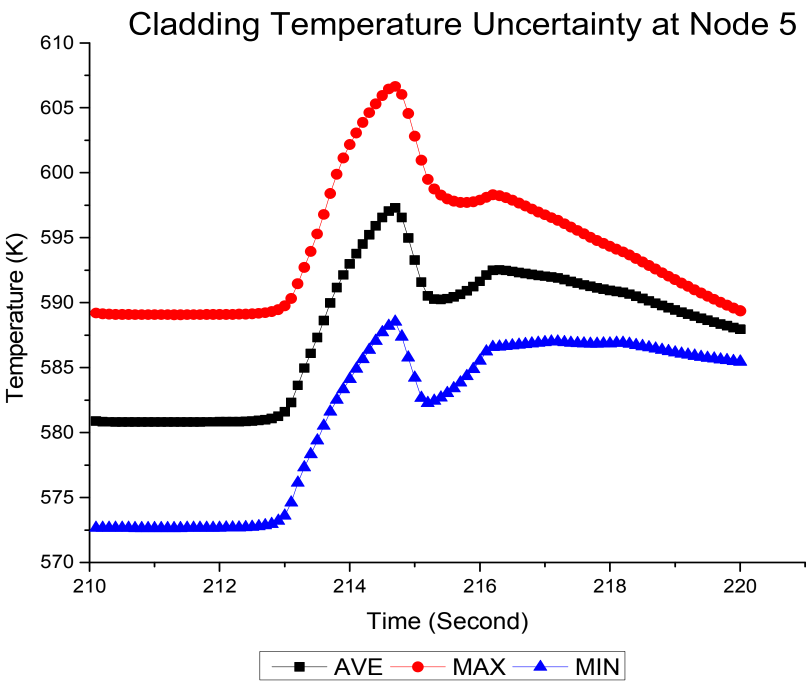


Figure 28 Uncertainty Band of Cladding Temperature at Node 5 in MSIVC Transient

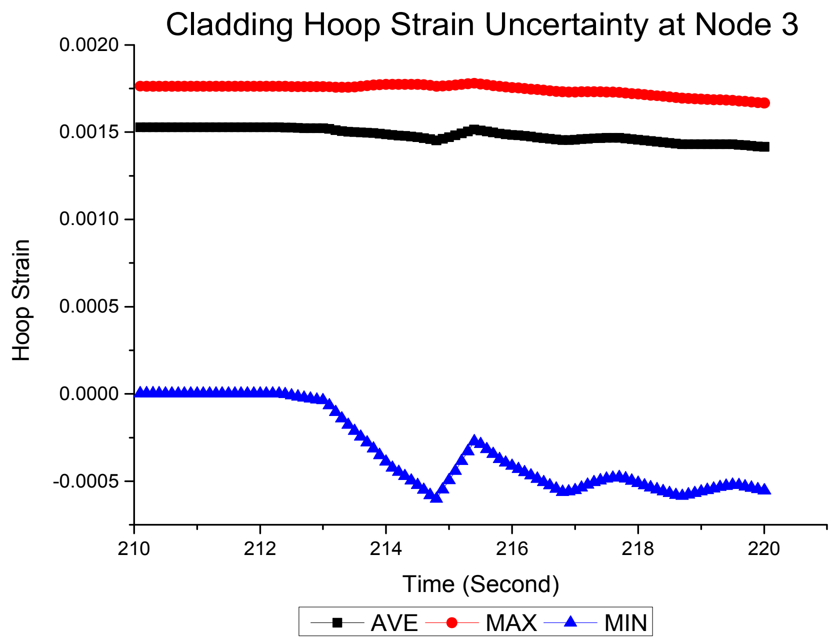


Figure 29 Uncertainty Band of Cladding Hoop Strain at Node 3 in MSIVC Transient

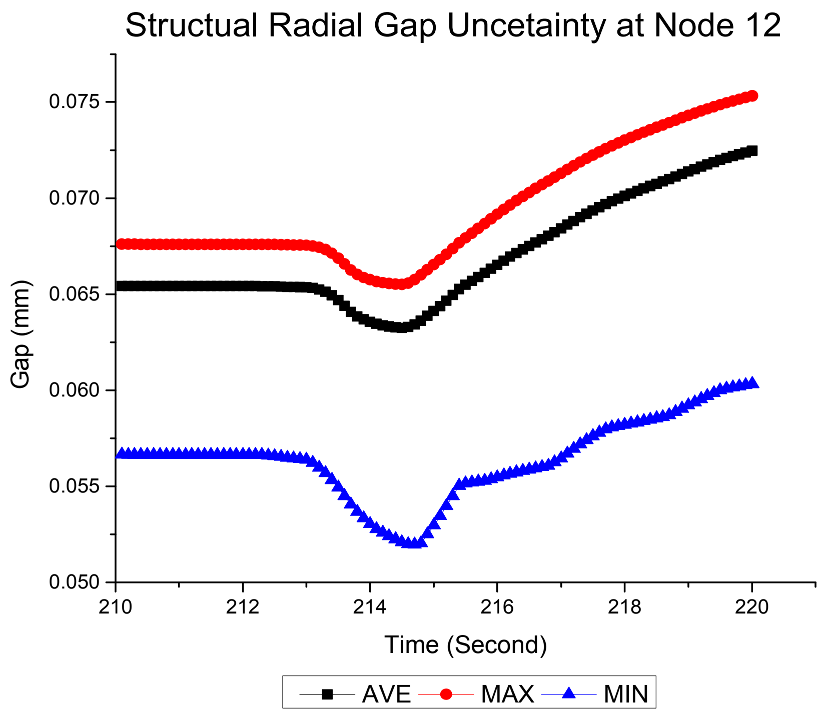


Figure 30 Uncertainty Band of Gap Thickness at Node 12 in MSIVC Transient

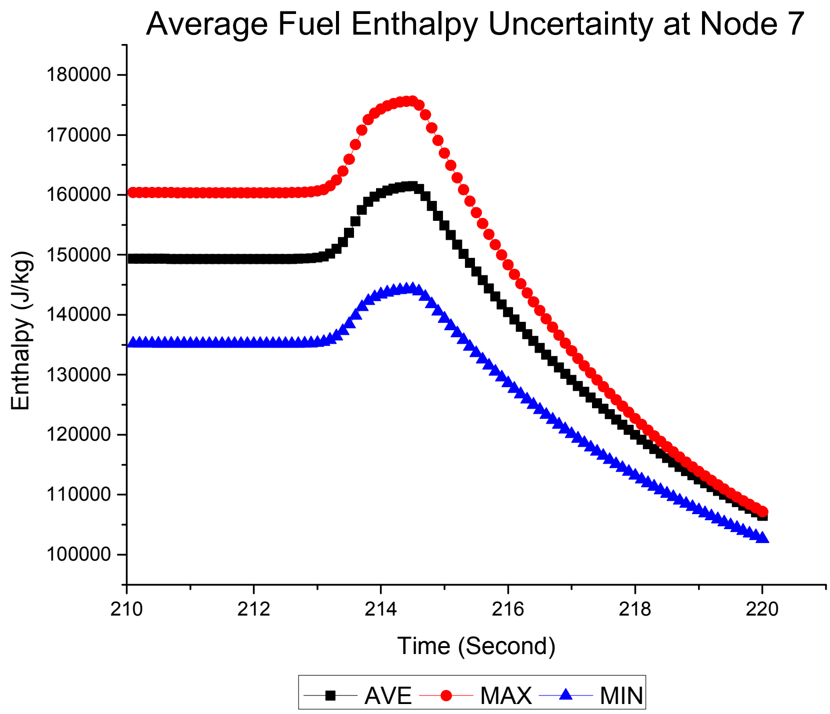


Figure 31 Uncertainty Band of Fuel Enthalpy at Node 7 in MSIVC Transient

4.2 Turbine Stop Valves Closure (TSVC)

Transient Assumption

In the hypothetical accident of TSVC, a 500 second steady state was performed to ensure all the parameters matched the operation conditions. At 500 second, the turbine tripped. As a result, 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 was performed and the power increased.

For conservative reason, on 6 safety relief valves opened with delayed time 0.4 second as the dome pressure reached to the setpoints 7.94 MPa. 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 5 listed the event sequences and important setpoints of the TRACE model during TSVC transient.

Table 5 Event Sequence of TSVC Transient

Time (sec)	Events	Notes
0~499	Steady state	Power 3030 MWt Feedwater flow rate 1641 kg/sec Feedwater temperature 488.7K Core inlet flow rate 11177.3 kg/sec Dome pressure 7.17MPa
500	TSVs start to close	Fully closure time 0.1 second
500.01	Reactor scram signal initiated	Signal initiated at TSVs 90% open
500.09	Reactor scram	Delayed time 0.08 second
500.1	TSVs fully closed	
501.77	Relief valves opened	Open at dome pressure 7.94MPa and close at dome pressure 7.63MPa
502.27	Reactor dome pressure peak value	Dome pressure 8.20 MPa
505	End of the analysis	

Analysis Data Results

The TSVs started to close at 500 second and fully closed at 500.1 second. Figure 32 shows that at the beginning of the transient, the steam flow dropped rapidly. The steam inside the reactor vessel was trapped; hence, the dome pressure increased. 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 32. 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 negative reactivity feedback would be decreasing with time. Hence, as shown in Figure 33, the core power kept increasing until the scram feedback dominated the power variation at 500.14 second.

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 34, the relief valves functioned when the dome pressure reached to 8.1 MPa. The steam flow rate increased again. 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 above, it is noticed that the reactor scram system can successfully inhibit positive reactivity which comes from the decreasing void fraction. In addition, even though there are only six relief valves functioned, the dome pressure can still be reduced sufficiently.

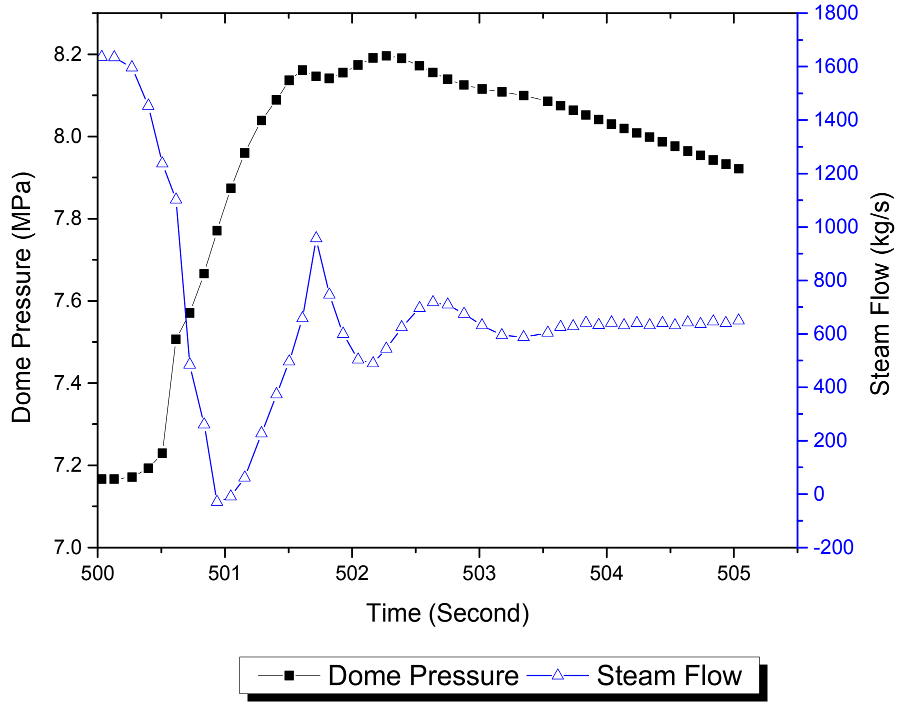


Figure 32 Variation of Dome Pressure and Steam Flow Rate in TSVC Transient

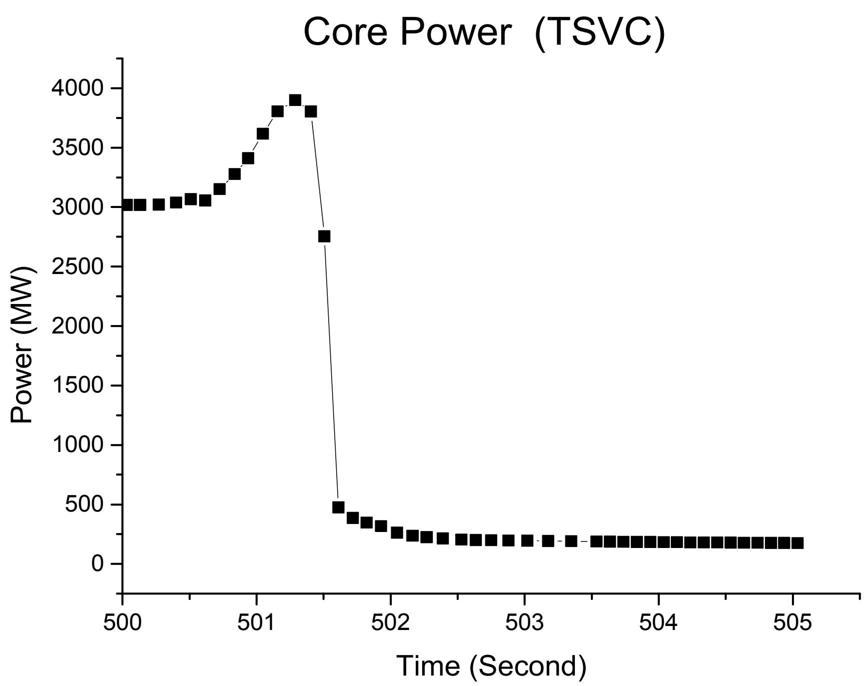


Figure 33 Variation of Core Power in TSVC Transient

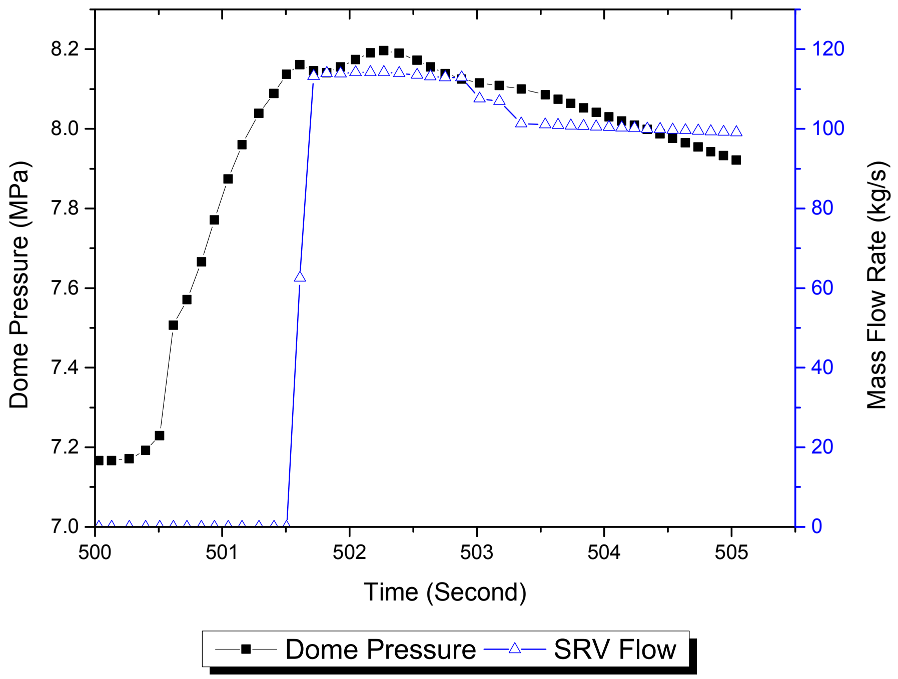


Figure 34 Relationship between Dome Pressure and Rerlief Valves Flow Rate

With the thermal hydraulic data results, the fuel rods properties of the TSVC transient can be calculated by FRAPTRAN code. As shown in Figure 35, the cladding temperature increased with power increasing. Due to the increasing temperature of the fuel rod, the gas inside would inflate and expand the cladding. On the other hand, the increasing dome pressure would squeeze the fuel cladding. As shown in Figure 36, at the beginning of the TSVC transient, the temperature increased greatly; as a result, the gap gas pressure inside cladding dominated the cladding hoop strain variation. After the reactor scram, the power dropped rapidly and the temperature decreased. The dome pressure outside cladding dominated the hoop strain variation. In addition to the cladding temperature and hoop strain, fuel pellet enthalpy is also an important criteria during the transient. Figure 37 shows that the maximum enthalpy in the TSVC transient was about 223512 J/kg, which was much lower than the criteria 710600 K/kg. Same as the MSIVC transient, the gap distance and the oxide thickness should also be checked in TSVC transient. Figure 38 shows that the oxide layer did not grow during the transient. There was no water-metal reaction in TSVC transient. Moreover, the fuel pellet surface expanded because of the increasing temperature at the beginning of the transient; as a result, the gap distance reduced before reactor scram. Once the reactor scrammed, fuel pellet shrank due to the decreasing temperature. The gap distance extended as shown in Figure39. Overall, the gap distance of all the positions were always larger than 0.06 mm, which means that there is no PCMI effect during the TSVC transient.

After the general analysis, uncertainty analysis data results of fuel rod properties are shown below. According to the Figure 35, position Node 8 was chosen the uncertainty band of cladding temperature in Figure 40. This figure shows that, with the manufacturing tolerance concern, the cladding temperature might reach to 627K during the power increasing transient. Further, according to Figure 36, Figure 41 shows that the uncertainty band of cladding hoop strain at Node 8 would increase up to 0.002, which was much lower than the criteria 0.01. Different with the variation in MSIVC transient, the cladding hoop strain in TSVC transient was always larger than 0. The cladding was not compressed. Nonetheless, the geometric uncertainty might influenced the deformation mechanism. As a result, Figure 42 shows the uncertainty band of gap distance at Node 8. From this figure, the gap distance was at least larger than 0.048 mm. The PCMI effect did not happen in this transient. Figure 43 shows the uncertainty of fuel enthalpy at Node 8. The peak value of the fuel enthalpy was about 25400 J/kg, which was much lower than the criteria. From the TRACE and FRAPTRAN data results mentioned above, it can be confirmed that both the reactor vessel and fuel rods kept good integrity during the TSVC transient.

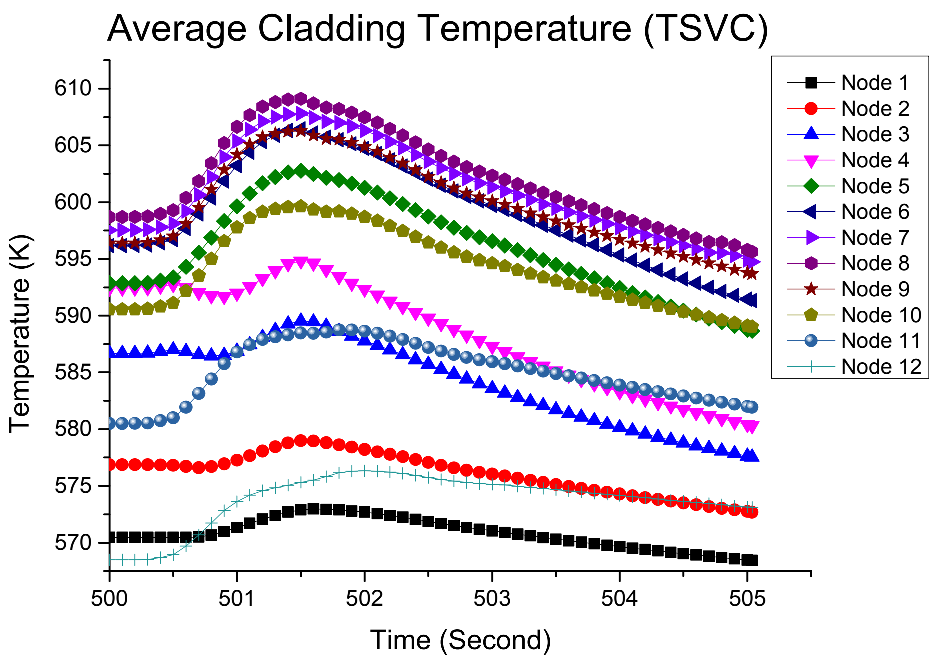


Figure 35 Cladding Temperature of FRAPTRAN Results in TSVC Transient

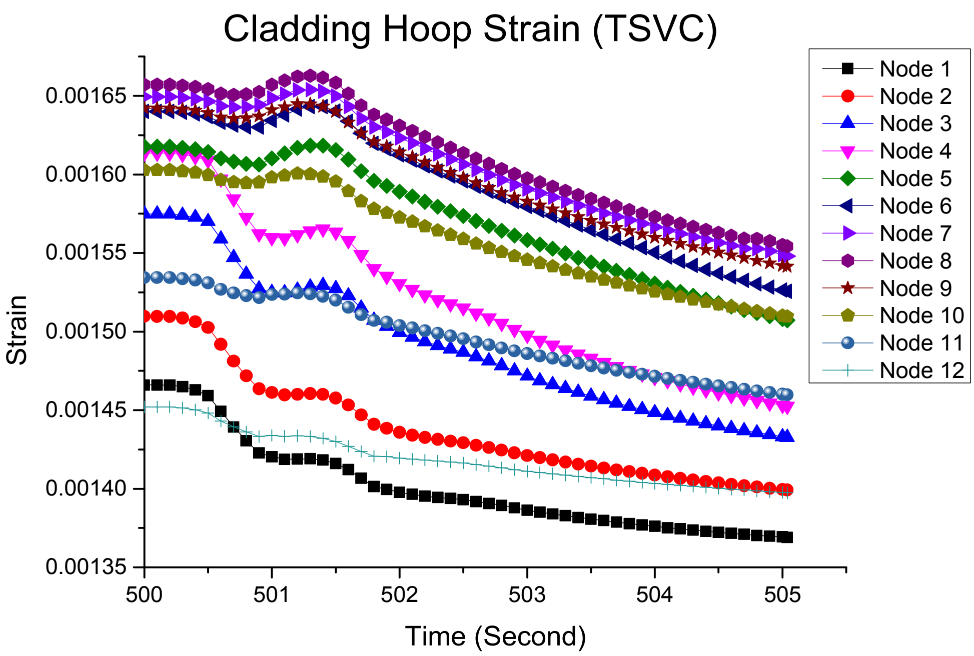


Figure 36 Cladding Hoop Strain of FRAPTRAN Results in TSVC Transient

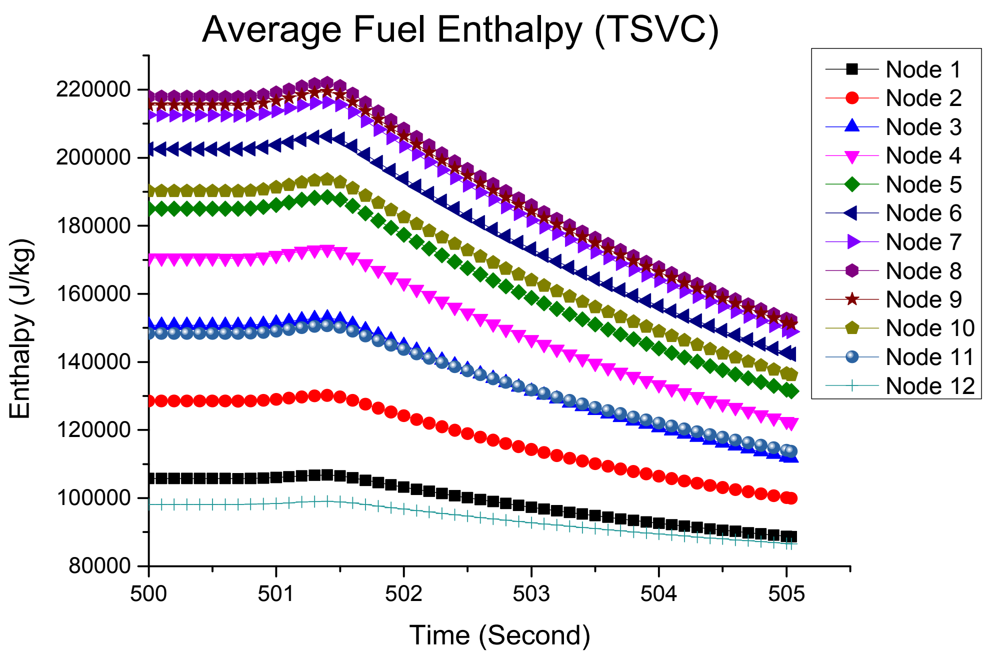


Figure 37 Fuel Enthalpy of FRAPTRAN Results in TSVC Transient

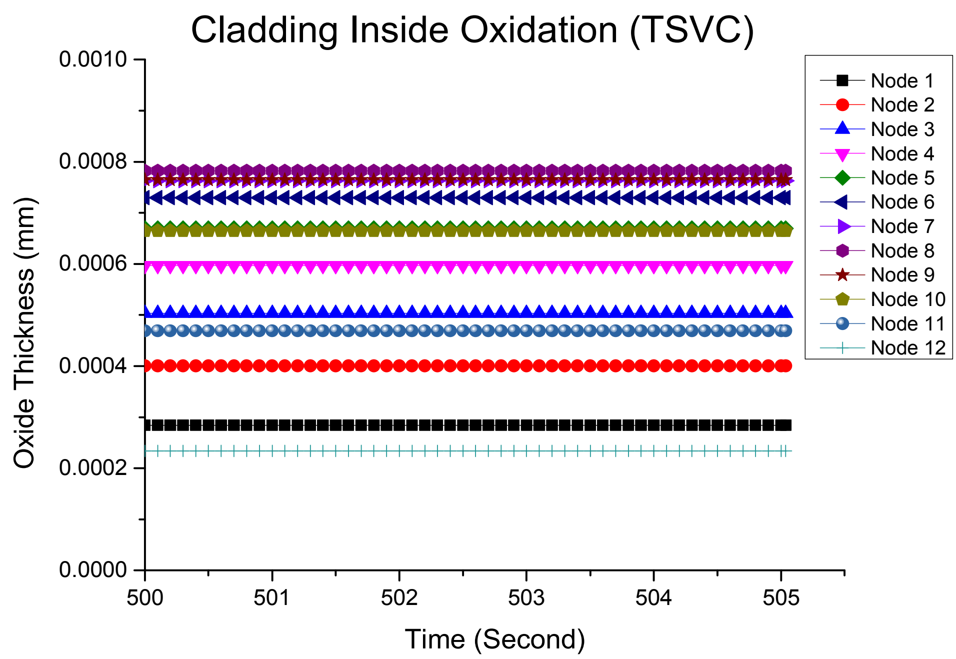


Figure 38 Oxidation Thickness of FRAPTRAN Results in TSVC Transient

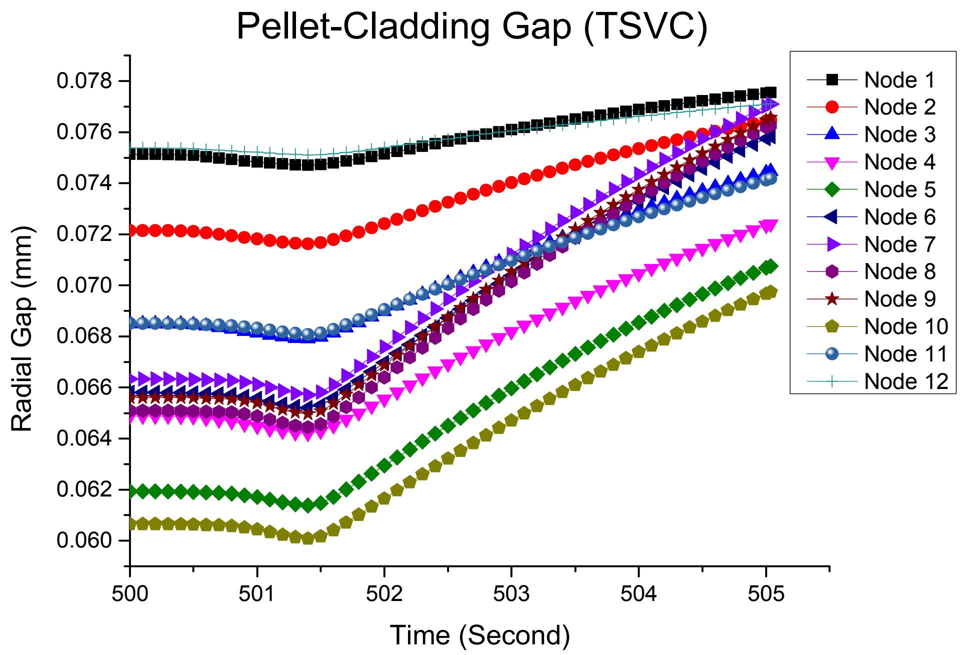


Figure 39 Gap Thickness of FRAPTRAN Results in TSVC Transient

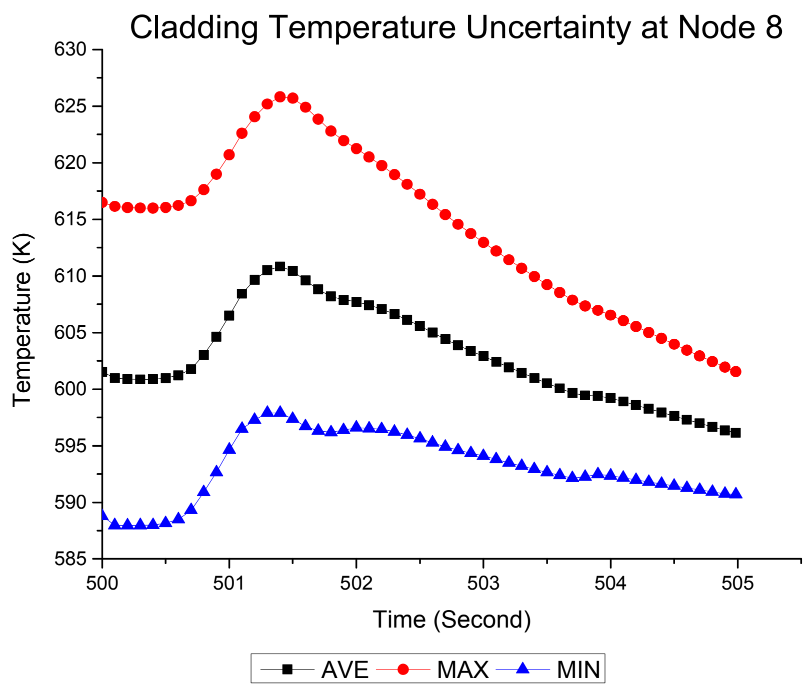


Figure 40 Uncertainty Band of Cladding Temperature at Node 8 in TSVC Transient

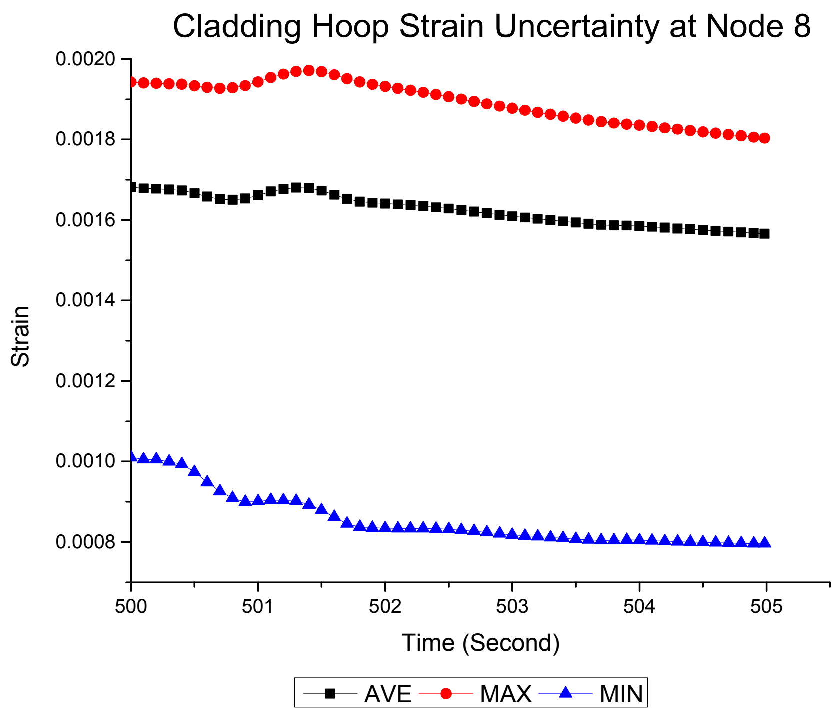


Figure 41 Uncertainty Band of Cladding Hoop Strain at Node 8 in TSVC Transient

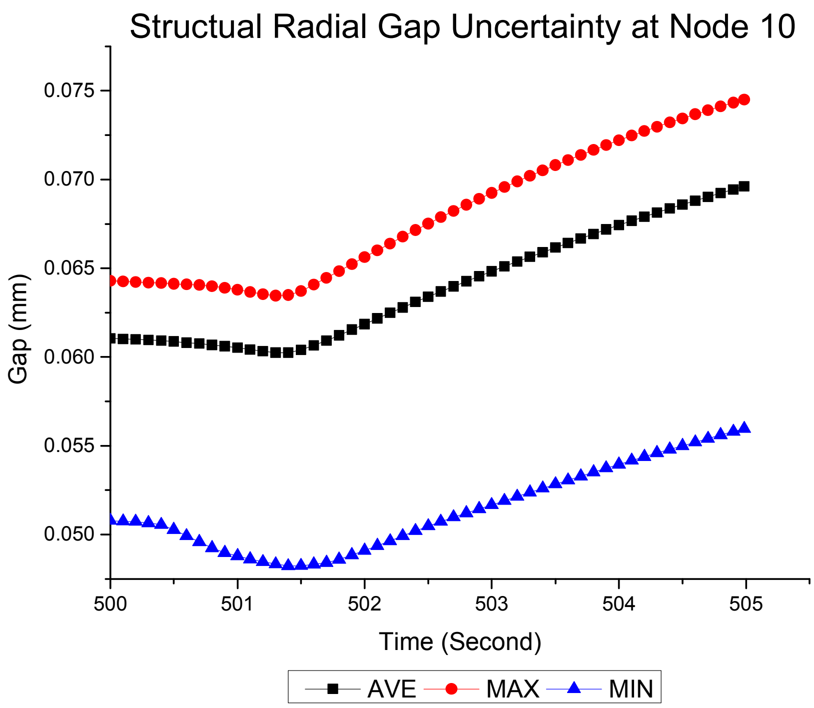


Figure 42 Uncertainty Band of Gap Thickness at Node 10 in TSVC Transient

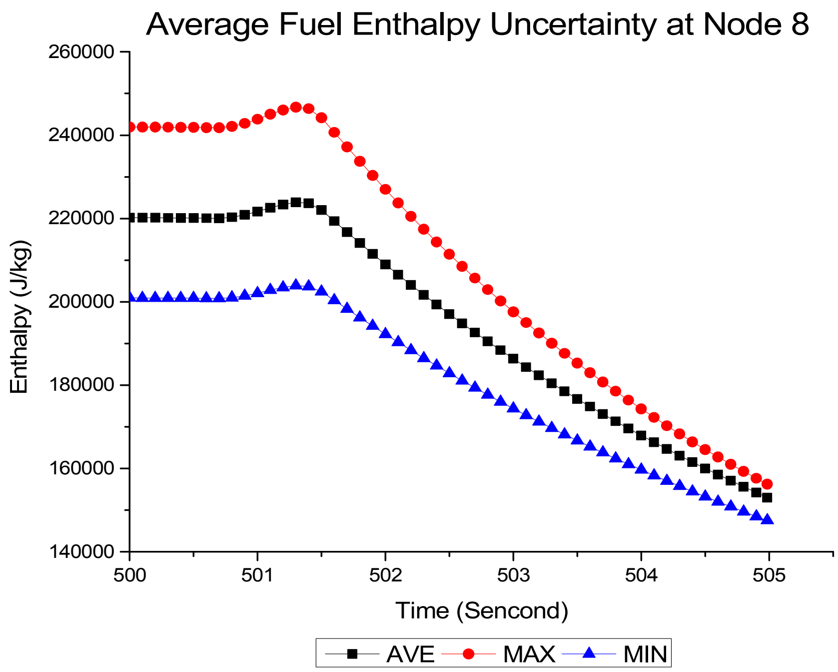


Figure 43 Uncertainty Band of Fuel Enthalpy at Node 8 in TSVC Transient

4.3 Turbine Control Valves Closure (TCVC)

Transient Assumption

Once the turbine control valves closed, the main steam line flow decreased immediately which increased the dome pressure. The positive void fraction reactivity feedback performed and the power increased. Different from the previous two cases, the recirculation pumps trip signal and the reactor scram signal were sent out immediately. However, due to the signal delayed time, the reactor would scram 0.07 second later and 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. Same as previous two cases, once the safety/relief valves open, the dome pressure decreased and the NPP was under controlled. Table 6 lists the important setpoints and transient events of the TCVC hypothetical accident.

Table 6 Event Sequence of TCVC Transient

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

Analysis Data Results

Figure 44 shows that once the turbine control valves closed, the steam flow rate dropped and the dome pressure increased immediately. As a result, the void fraction inside the reactor core decreased which caused a positive reactivity feedback. The power increased as shown in Figure 45. 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. After 210.5 second, the power increased because the positive void fraction reactivity feedback dominated the power. However, at 211.2 second, 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. The pressure was 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 46 shows that the feedeater flow rate and the recirculation flow rate decreased. After 212 seconds, the feedwater flow rate kept constant because the water level of the reactor vessel was stable at this value.

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.

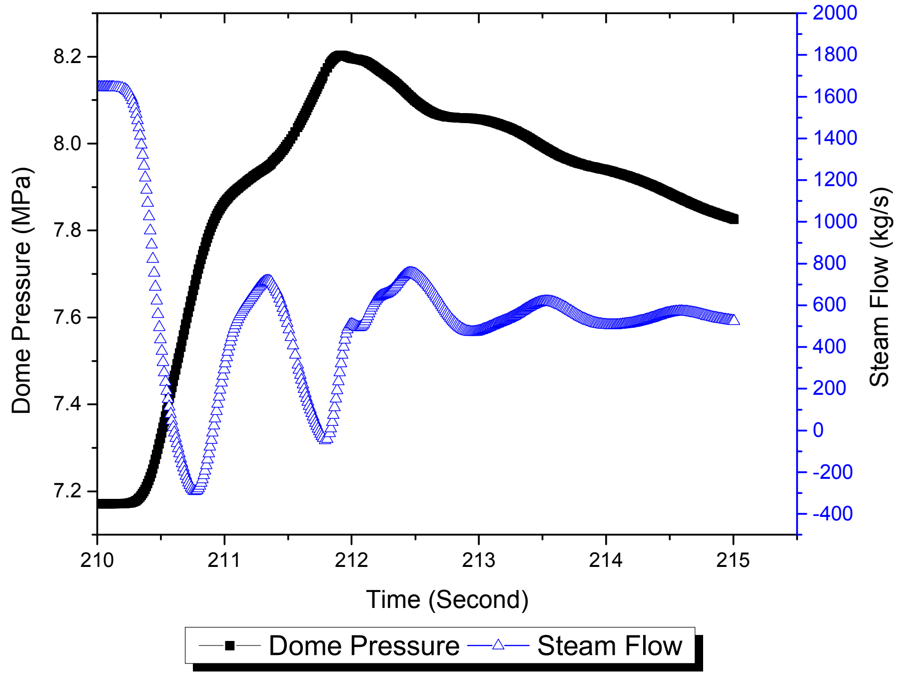


Figure 44 Variation of Dome Presssure and Steam Flow Rate in TCVC Transient

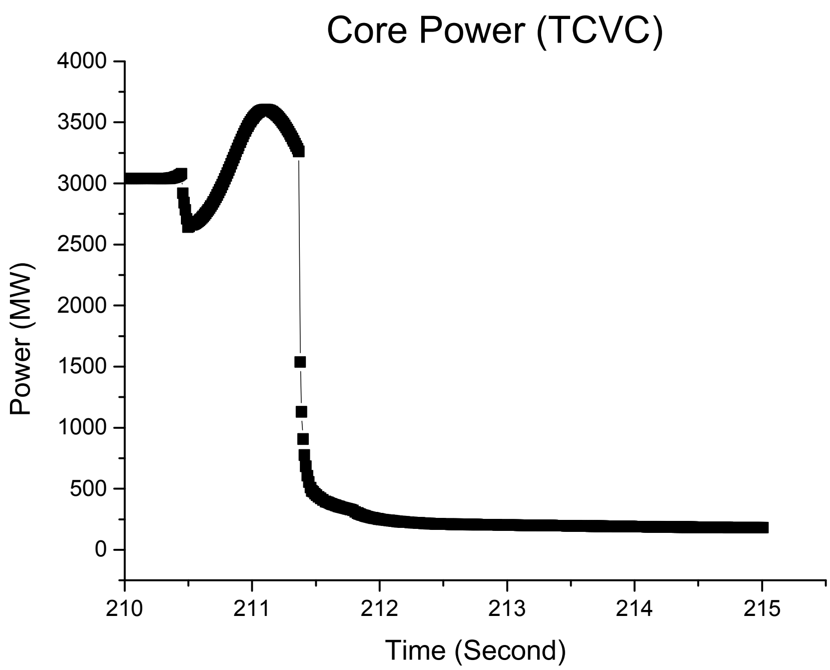


Figure 45 Variation of Core Power in TCVC Transient

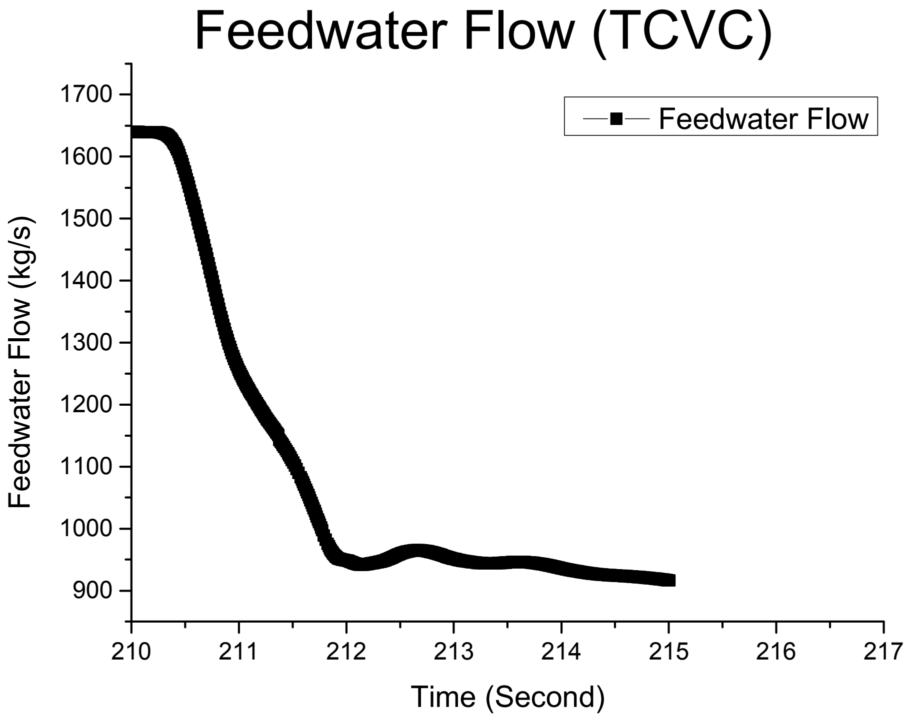


Figure 46 Variation of Feedwater Flow Rate in TCVC Transient

With the thermal hydraulic data results, the fuel rods properties of the TCVC transient can be calculated by FRAPTRAN code. As shown in Figure 47, once the TCVs started to close, the power increased and as a result, the cladding temperature of all positions raised. However, the reactor scrammed immediately as the TCVs start to close. The period of power increasing was shorter than that of the previous two cases. Hence, the peak cladding temperature would be about 588K which was lower than that of previous two cases. On the other hand, because the cladding temperature did not increase a lot, the cladding hoop strain shrank since the beginning of the transient as shown in Figure 48. This variation indicates that the cladding would not expand and rupture but the PCMI effect might happen. As a result, the gap distance variation between fuel pellet and cladding should be examined. Figure 49 shows that at the beginning of the transient, the gap distance decreased because of the increasing power and decreasing hoop strain. After 211.5 second, the power dropped rapidly and hence the fuel pellet would no longer expand. Further, with the opening of the safety/relief valves after 211.5 second, the dome pressure decreased. Based on the above two points, the gap distance increased. Figure 50 shows the variation of fuel enthalpy during the TCVC transient. Due to the power increasing being controlled rapidly, the fuel enthalpy did not increase a lot. The peak value of enthalpy is about 15500 J/kg, which is much lower than the criteria regulated by U.S. NRC. In addition, Figure 51 shows the variation of the oxide thickness. From this figure, it is obviously that the oxidation layer had not been produced.

After the general analysis, uncertainty analysis data results of fuel rod properties are shown below. According to the results of Figure 47, the highest peak cladding temperature was appeared at position Node 4. As a result, Figure 52 shows the uncertainty band of cladding temperature at Node 4. With the manufacturing tolerance, the peak cladding temperature might reached to 598K. After the reactor scram, the power decreased and hence the deviation of the uncertainty band reduced. According to the results of Figure 48, Figure 53 shows the uncertainty band of cladding hoop strain at Node 5. From this figure, the peak cladding hoop strain would reach to 0.0018, which was still much lower than the criteria 0.01. Further, the negative deviation of hoop strain was larger than 0, which means that the cladding would not be squeezed. However, the fabrication uncertainty might change the variation of the fuel swelling. The PCMI effect would still possibly happen. As a result, according to Figure 49, Figure 54 shows the uncertainty band of gap distance at Node 12. From this figure, the gap distance was always larger than 0.056 mm. The PCMI effect would not happen in this transient. Figure 55 shows the uncertainty band of fuel enthalpy at position Node 8. The variation of enthalpy was very similar to that of cladding temperature. The larger deviation was appeared in power increasing period. After the reactor scram, the deviation would reduce. In spite of considering manufacturing tolerance, the peak fuel enthalpy did not exceed the criteria 170 cal/g. Based on these data results mentioned above, the structure of the power plant was safe and the fuel rods still maintained good integrity.

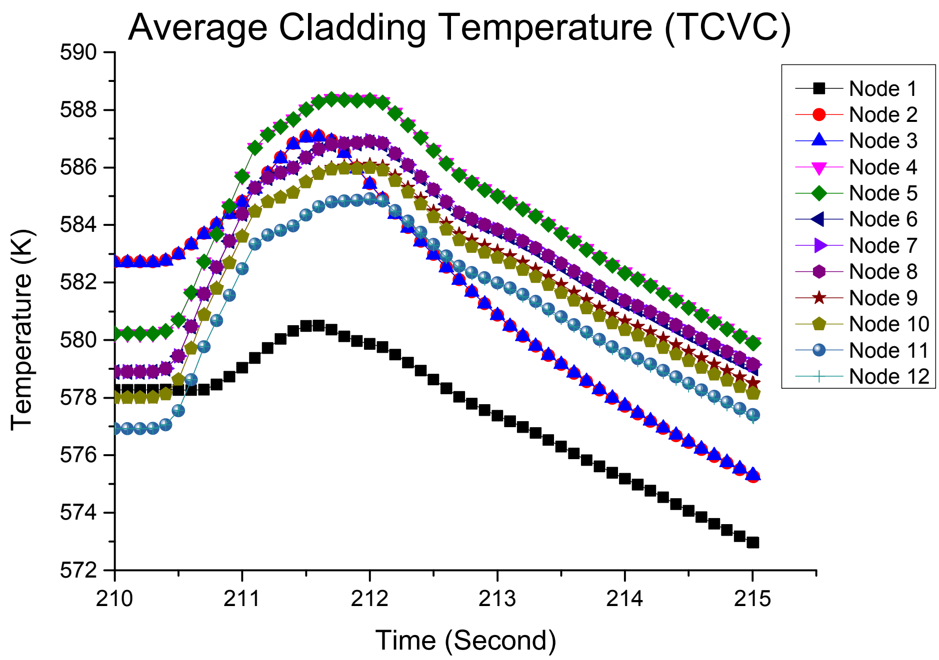


Figure 47 Cladding Temperature of FRAPTRAN Results in TCVC Transient

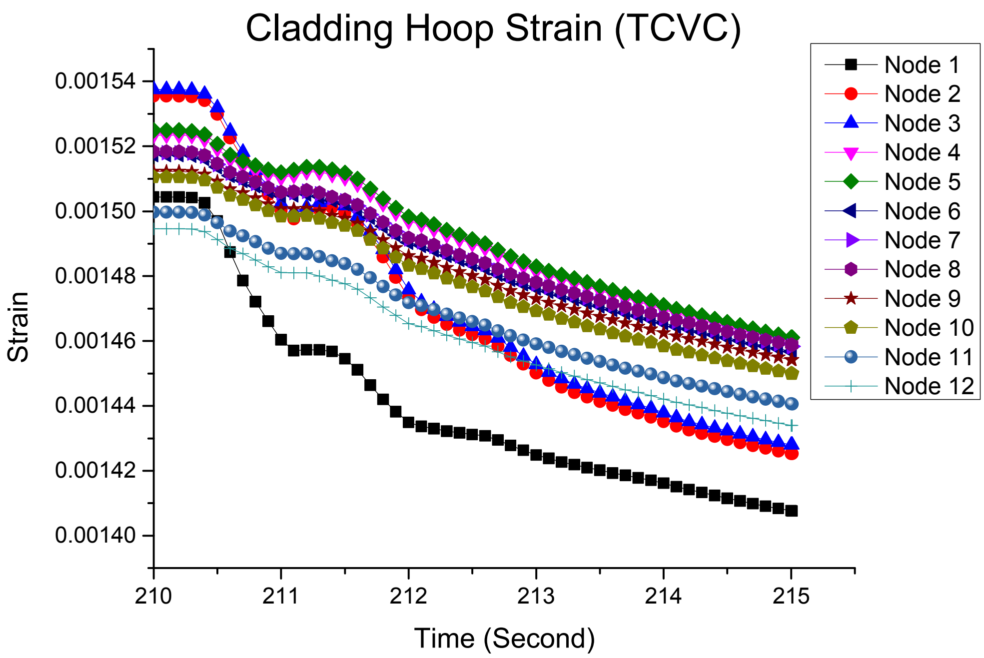


Figure 48 Cladding Hoop Strain of FRAPTRAN Results in TCVC Transient

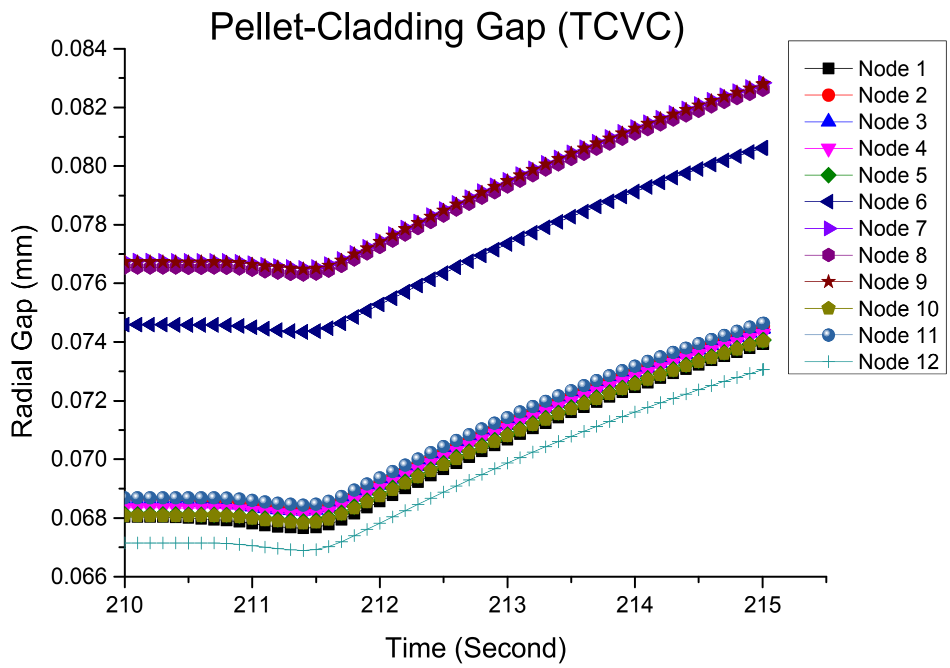


Figure 49 Gap Thickness of FRAPTRAN Results in TCVC Transient

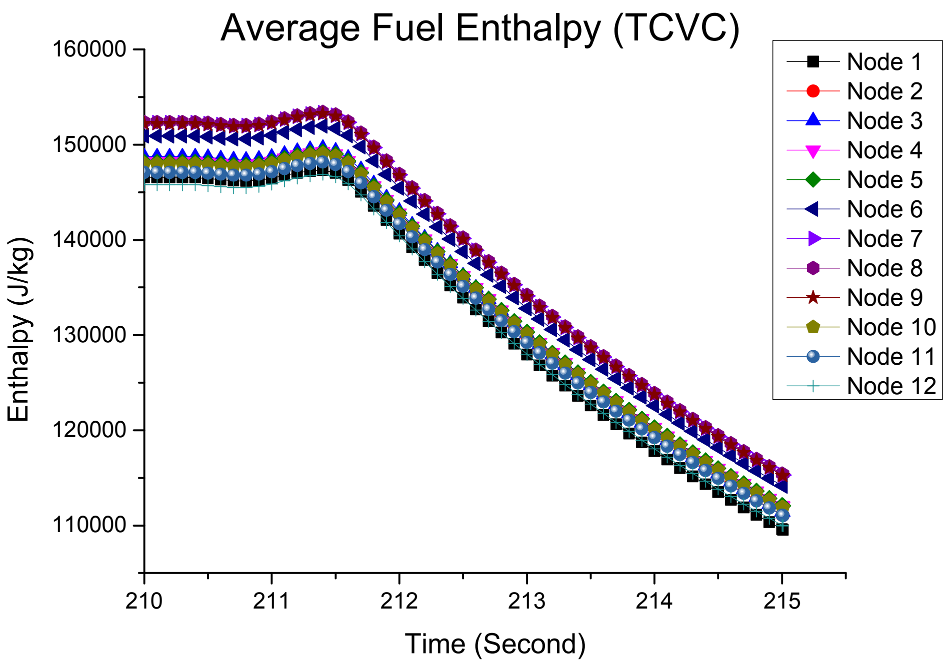


Figure 50 Fuel Enthalpy of FRAPTRAN Results in TCVC Transient

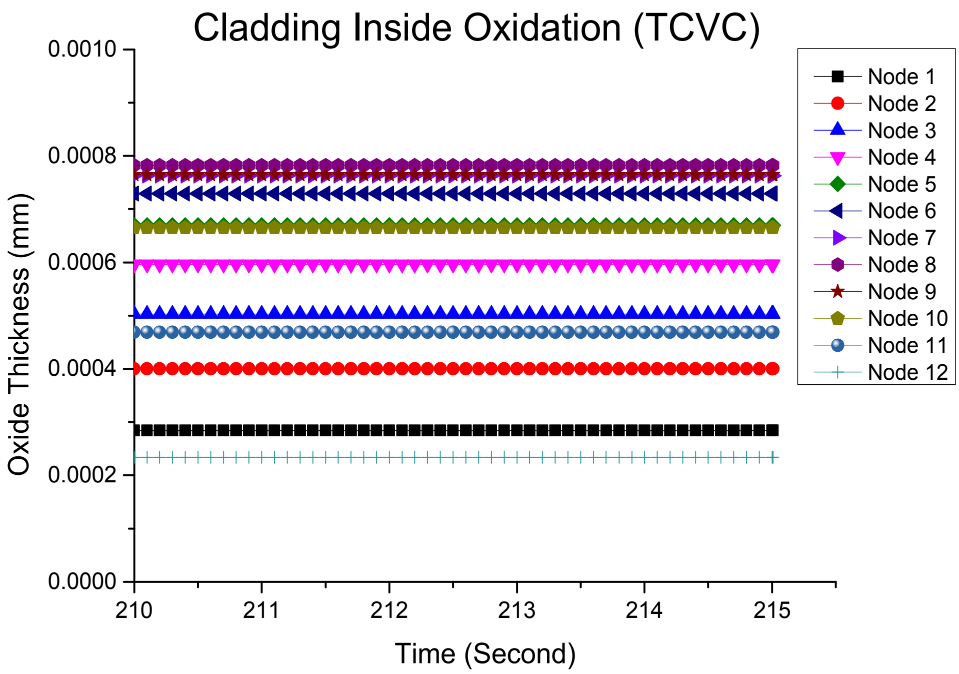


Figure 51 Oxidation Thickness of FRAPTRAN Results in TCVC Transient

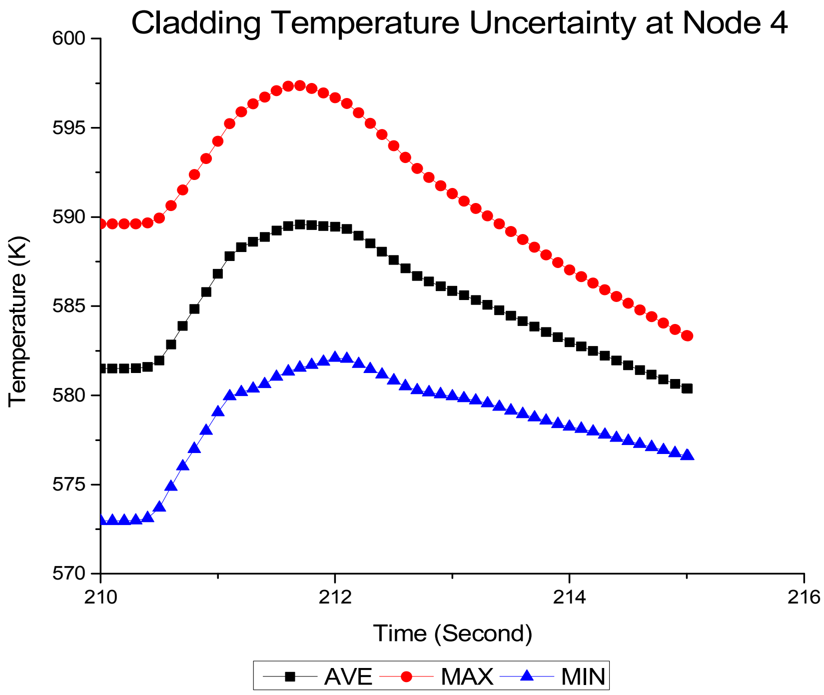


Figure 52 Uncertainty Band of Cladding Temperature at Node 4 in TCVC Transient

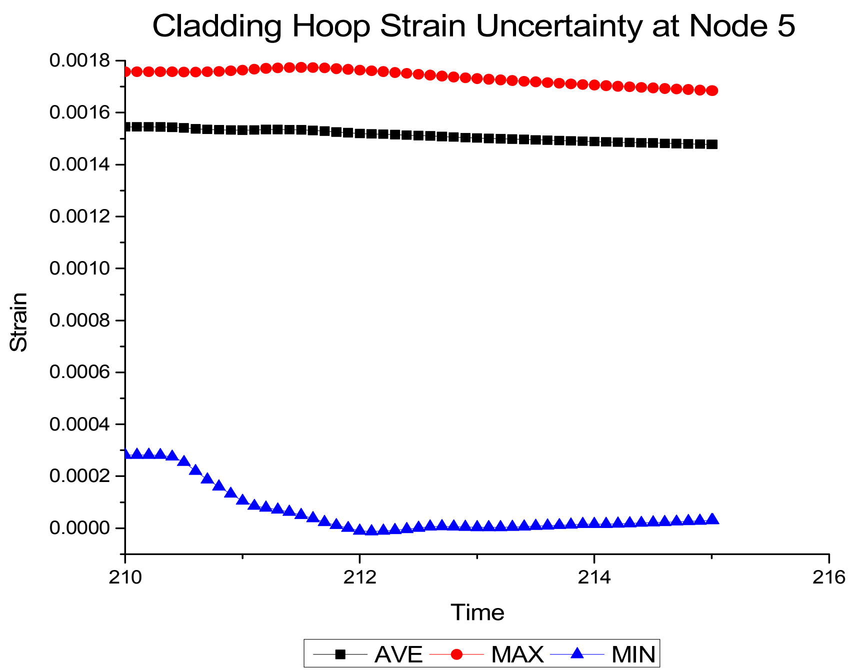


Figure 53 Uncertainty Band of Cladding Hoop Strain at Node 5 in TCVC Transient

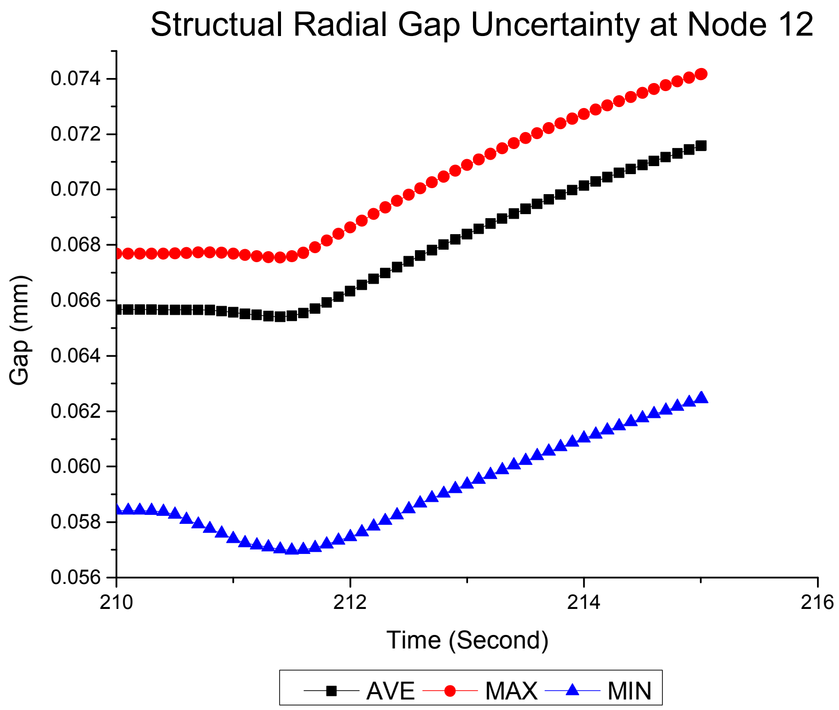


Figure 54 Uncertainty Band of Gap Thickness at Node 12 in TCVC Transient

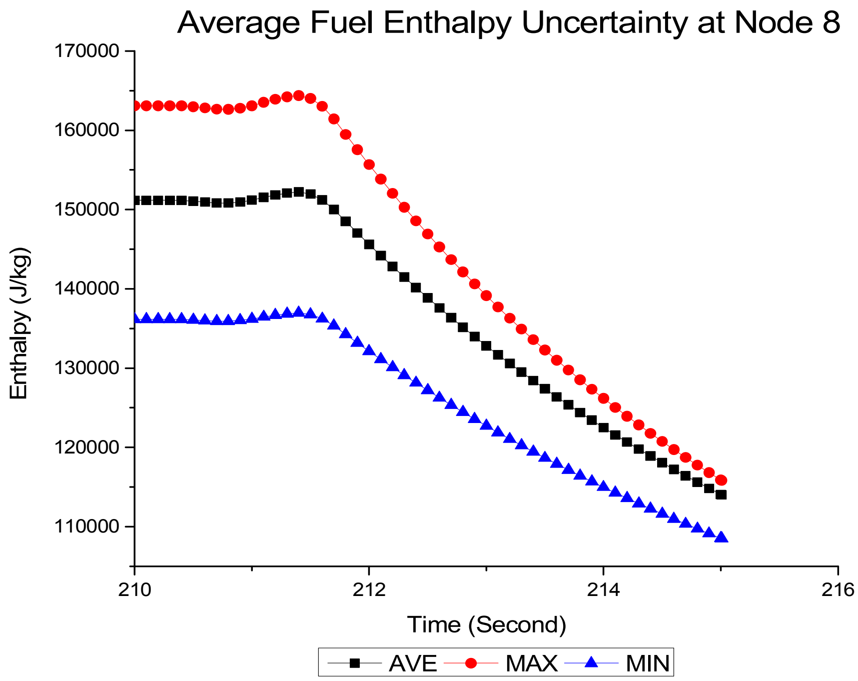


Figure 55 Uncertainty Band of Fuel Enthalpy at Node 8 in TCVC Transient

4.4 <u>Transient Comparison</u>

From these data results, it is obvious that the variation of power, dome pressure and fuel rods properties are familiar for these three overpressurization transient. However, due to the difference of the valves closure time, peak values of dome pressure and other fuel rods properties are quite different. Table 7 lists the comparison of the important parameters for these three hypothetical accident. From this table, due to the longer power increasing period, the range of power increasing in MSIVC is much larger than that in the other two cases. Further, in the MSIVC transient, it was assumed that the relief valves failed. There were only 11 safety valves functioned. As a result, the discharging of steam was less and the dome pressure of the MSIVC transient was larger than the other two cases. In addition, from Table 7, it should be noticed that the time points of peak dome pressure were later than that of the safety/relief valves opening. It means that the discharging rates of the steam through these valves were less than the generating rates of the steam inside the reactor vessel.

For the fuel rod properties, Figure 56 shows the comparisons of peak value time of the three criteria. From this figure, it should be noticed that the timepoints of peak fuel properties would delay for few seconds after that of peak power. For example, in the TSVC and TCVC transients, the fuel enthalpy which was influenced mainly by the fuel temperature would reach to the peak value first because of the increasing reactivity and power. After that, the increasing power heated the gas inside the gap of the fuel rods. The gas inflated and expanded the cladding. As a result, the cladding hoop strain increased. Then, the gas transferred the heat to the cladding. Hence, the cladding temperature reached to the peak value the last. However, in the MSIVC transient, the dome pressure increased a lot. The gas inflation inside the gap was countervailed by the outside coolant pressure at the beginning of the transient. Hence, the cladding hoop strain reached to the peak value the last, which was different from the variation of TSVC and TCVC transient.

Table 7 Comparison of the Important Parameters for Three Hypothetical Accident

Events	MSIVC	TSVC	TCVC
Valves closure time (Sec)	3	0.243	0.16
Maximum power (MWt)	6933	3900	3608
Maximum power time (Sec)	3.62	1.29	1.10
Peak value of dome pressure (MPa)	8.82	8.20	8.20
Maximum dome pressure time (Sec)	4.77	2.27	1.90
Reactor scram time (Sec)	3.25	0.24	0.08
Peak cladding temperature (K)	606.70	625.88	597.43
Maximum cladding temperature time (Sec)	4.67	1.41	1.69
Maximum cladding hoop strain	0.00178	0.00197	0.00177
Maximum cladding hoop strain time (Sec)	5.41	1.4	1.51
Maximum fuel enthalpy (J/kg)	1.70E+05	2.47E+05	1.64E+05
Maximum fuel enthalpy time (Sec)	4.38	1.30	1.40

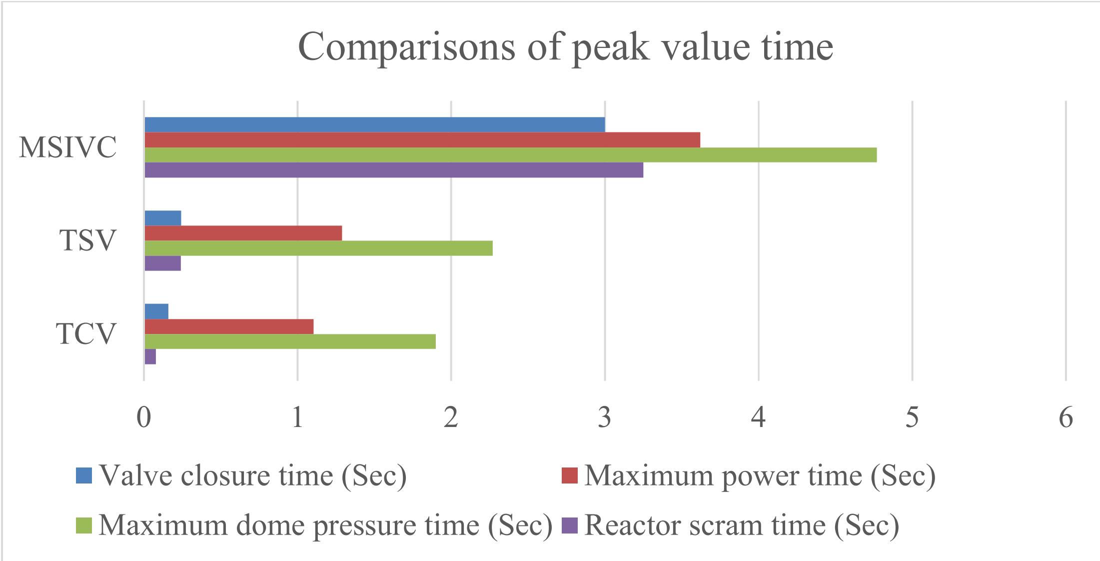


Figure 56 Comparisons of Peak Value Time

5. CONCLUSIONS

From these analysis data, Kuosheng NPP will be safe during the overpressurization transient after SPU. In spite that turbine bypass valve failed, the dome pressure would not exceed the criteria regulated by ASME if certain amounts of safety/relief valves opened. Further, the closure time of valves in this research were shorter than that in reality. The hypothetical transient conditions were more severe. However, the reactor scram setting and other safety mechanism could still maintain the integrity of the NPP and the fuel rods. On the other hand, for the uncertainty analysis, the manufacturing tolerance might cause 1.7% of cladding temperature deviation, 14% of cladding hoop strain deviation and 10% of fuel enthalpy deviation. Nevertheless, even concerning the uncertainty deviation, the fuel rod properties would not exceed the criteria.

In this research, a method, which combines thermal hydraulic and fuel rod properties analysis, was successfully built up. The researchers can not only obtain the thermal hydraulic data of the NPP in transient, but also can observe the fuel rods properties through these thermal hydraulic results. Further, with the SNAP interface, power plant models of TRACE and FRAPTRAN can be built up more efficiently. The visible control blocks and thermal hydraulic components help researchers comprehend the control logic and the power plant model more easily. However, due to the restriction of the program, data transfer from TRACE to FRAPTRAN was not automatically in this research. It needs a lot of human resources to transfer thermal hydraulic data results to the FRAPTRAN input deck. Hope that in the near future, a program which can transfer data results from TRACE to FRAPTRAN codes will be developed.

With the advantages of DAKOTA code setting in SNAP interface, an uncertainty analysis methodology was also built up in this research. With this method, the manufacturing tolerance influence for the fuel rods in transient could be concerned. However, due to the restriction of data collection, there were only six fuel rod geometric parameters discussed in this research. Hope that in the researches to come, the fabrication uncertainty information can be collected more completely. Even more, to gather a more convincing data results, the uncertainty of thermal hydraulic components should be collected and concerned in the future researches.

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K.Tien, NRC Project Manager

11. ABSTRACT (200 words or less)
After the MUR (measurement uncertainty recapture) power uprates, Kuosheng Nuclear Power Plant has uprated the power to 2943 MWt. Recently, Taiwan Power Company is concerned in SPU (stretch power uprated) plan and uprates the power to 3030 MWt, which is 104.7% of the designed power. Before the stretch power uprates, several transient analysis should be done for ensuring that the power plant could maintain stability in high power operating conditions. The advanced thermal hydraulic analysis code TRACE which is conducted by U.S. NRC was applied for overpressurization transient including main steam line isolation valves closure, turbine stop valves closure and turbine control valves closure. This closure of valves increased the dome pressure; as a result, the void fraction inside the reactor core decreased. The decline of the void fraction will increase the reactivity feedback; hence, the power increased rapidly until the reactor scram. Further, with safety relief valves open, the dome pressure would not exceed the criteria regulated by ASME. In addition, to cover the insufficiency of thermal hydraulic code, fuel rod transient analysis code FRAPTRAN was applied. To perform fuel rod transient analysis, thermal information from TRACE code would be entered into FRAPTRAN with power history. In addition, DAKOTA code was applied to concern the influence of fuel rod manufacturing tolerance. With uncertainty bands from DAKOTA analysis, the criteria could be judged with more confidence. This research successfully established a procedure of thermal hydraulic and fuel rod property analysis. Further, with SNAP interface, the FRAPTRAN and DAKOTA were combined successfully to perform the uncertainty analysis.

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TRACE/FRAPTRAN/DAKOTA
Best Estimate plus Uncertainty (BEPU)
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Kuosheng Nuclear Power Plant (NPP)

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