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Fuel Rod Behavior and Uncertainty Analysis by FRAPTRAN/TRACE/DAKOTA Code in Maanshan LBLOCA

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ABSTRACT

In this study, the FRAPTRAN and TRACE code were used to evaluate the fuel rod transient behavior during a postulated LBLOCA in Maanshan (3-loops PWR) Nuclear Power Plant(NPP). There were three main steps in this research. The first step was the LBLOCA analysis for Maanshan NPP by TRACE code. The analysis results were benchmarked and compared with Maanshan FSAR data. In second step, the geometry data of the fuel rod and the results from TRACE analysis (e.g. fuel rod power, coolant pressure, heat transfer coefficient) were input into FRAPTRAN to analyze the reliability of fuel rod. Then, it used FRAPTRAN to calculate the response of a single fuel rod transient behavior during LBLOCA. FRAPTRAN can obtain the detail mechanical property of fuel rod (e.g. cladding temperature, hoop stress/strain, gap pressure, and oxide thickness of cladding). After all, uncertainty analysis was considered in this study. The several parameters of fuel rod, such as fabrication and boundary conditions, were quantized and sampled by the DAKOTA uncertainty code.

FOREWORD

The US 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 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) which includes the development and maintenance of TRACE. To meet this responsibility, the TRACE model of Maanshan NPP has been built. In this report, the TRACE and FRAPTRAN code were used to evaluate the fuel rod transient behavior during a postulated LBLOCA in Maanshan Nuclear Power Plant.

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EXECUTIVE SUMMARY

An agreement in 2004 which includes the development and maintenance of TRACE has been signed between Taiwan and USA on CAMP. INER is the organization in Taiwan responsible for applying TRACE to thermal hydraulic safety analysis in order to provide users' experiences and development suggestions. To fulfill this responsibility, the TRACE model of Maanshan NPP is developed by INER.

According to the user manual, TRACE is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool. 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. On the whole TRACE provides greater simulation capability than the previous codes, especially for events like LOCA.

Fuel Rod Analysis Program Transient (FRAPTRAN) is a FORTRAN language computer code which was developed by Pacific Northwest National Laboratory, PNNL. The main purpose of this code is to calculate the response of a single fuel rod transient behavior in Light Water Reactors (LWR) during operational transients or hypothetical accidents, such as Reactivity Accidents (RIA) or LOCA, up to burnup level of 62 GWd/MTU. FRAPTRAN calculates the fuel and cladding temperatures, cladding strain and stress, and plenum gas pressure at different time for given power and coolant conditions.

The Maanshan NPP operated by Taiwan Power Company (TPC) is the only Westinghouse-PWR in Taiwan. The rated core thermal power is 2775 MW. The reactor coolant system has three loops, each of which includes a reactor coolant pump and a steam generator. The pressurizer is connected to the hot-leg piping in loop 2. The main components of Maanshan TRACE model include the pressure vessel, pressurizer, steam generators, steam piping in the secondary side (including four sets of steam dump and vent valves), the steam dump system, accumulators, and safety injection of emergency core cooling system (ECCS). The pressure vessel is divided into 12 levels in the axial direction, two rings in the radial direction (internal and external rings) and six equal azimuthally sectors in the "θ" direction. The control rod conduit connects the 12th and 7th layers of the vessel from end to end. The fuel region is between the third and sixth layers, and heat conductors are added onto these structures to simulate the reactor core.

In this study, the FRAPTRAN and TRACE code were used to evaluate the fuel rod transient behavior during a postulated LBLOCA in Maanshan (3-loops PWR) nuclear power plant. There were three main steps in this research. The first step was the LBLOCA analysis for Maanshan NPP by TRACE code. The analysis results were benchmarked and compared with Maanshan FSAR data. In second step, it used FRAPTRAN to calculate the response of a single fuel rod transient behavior during LBLOCA. After all, uncertainty analysis was considered in this study. The several parameters of fuel rod, such as fabrication and boundary conditions, were quantized and sampled by the DAKOTA uncertainty code.

In summary, thermal-hydraulic analytical results indicate that the Maanshan TRACE model predicts the behaviors of important plant parameters in consistent trends with the FSAR data. In FRAPTRAN analysis, the hoop stress was about 18 MPa for the fuel rod simulation, which are all within the safety operation range. Based on the calculation results of plenum pressure and the gap thickness, there was no cladding ballooning or failure in the present case. The cladding temperatures and the fuel centerline temperatures were all below the criteria requirement. Considering uncertainty analysis, the uncertainty band is formed by the 59 calculations. The minimum and maximum PCT are 842K and 892K respectively. The uncertainty analysis shows that the PCTs were all below the criteria requirement. The maximum hoop stress was about 37 MPa and the maximum hoop strain was 0.006 for the fuel rod simulation, which are all within the safety operation range.

ABBREVIATIONS

CAMP	Code Applications and Maintenance Program
DAKOTA	Design Analysis Kit for Optimization and TerascaleApplication
DBAs	Design Basis Accidents
ECCS	Emergency Core Cooling System
FRAPTRAN	Fuel Rod Analysis Program Transient
FSAR	Final Safety Analysis Report
INER	Institute of Nuclear Energy Research Atomic Energy Council, R.O.C.
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LWR	Light Water Reactors
NRC	Nuclear Regulatory Commission
NPP	Nuclear Power Plant
PCT	Peak Cladding Temperature
RIA	Reactivity Accidents
SNAP	Symbolic Nuclear Analysis Package
TPC	Taiwan Power Company
TRACE	TRAC/RELAP Advanced Computational Engine

1. INTRODUCTION

The Maanshan Nuclear Power Plant (NPP) operated by Taiwan Power Company (TPC) has two Westinghouse PWR units. The rated core thermal power of each unit is 2775 MW. The reactor coolant system has three loops, each of which includes a reactor coolant pump and a steam generator. The pressurizer is connected to the hot-leg piping in loop 2. The main components of Maanshan TRACE model include the pressure vessel, pressurizer, steam generators, steam piping in the secondary side (including four sets of steam dump and vent valves), the steam dump system, accumulators, and safety injection of Emergency Core Cooling System (ECCS).

Loss of Coolant Accident (LOCA) is one of the important Design Basis Accidents (DBAs) in light water reactors, and the Large Break LOCA (LBLOCA) is the most serious one. The LBLOCA is a double-ended guillotine break of the largest primary system piping and is the limiting condition for ECCS requirements. In this study, a LBLOCA was defined as a rupture in Maanshan NPP cold-leg with a total cross sectional area. The break was set in loop 1, which is the one out of two loops that doesn't have a pressurizer. Referring to 10CFR50.46[1], the current LOCA criteria are briefly listed below:

1) The peak clad temperature shall not exceed 1477.6 K;
2) The maximum thickness of the cladding oxidation shall not exceed 17% of the clad thickness;
3) The maximum hydrogen generation by cladding oxidation is no more than 1% of the total amount;
4) The maintenance of cooling geometry;
5) The maintenance of long term cooling.

Nowadays, it is becoming a trend to evaluate the NPP safety involving several disciplines, such as thermal hydraulic, thermal mechanics, and reactor physics[2, 3]. In this paper, the previous results of LBLOCA studied with TRACE were used as the input boundary conditions for FRAPTRAN. By using this fuel rod analysis program, the fuel behaviors during LBLOCA transients in Maanshan NPP can be learned more comprehensively.

2. METHODOLOGY

2.1 Method

In this study, the FRAPTRAN and TRACE code were used to evaluate the fuel rod transient behavior during a postulated LBLOCA in Maanshan NPP. In addition, the uncertainty analysis by DAKOTA code was also discussed in the fuel properties. Figure 1 shows the flowchart of combining FRAPTRAN and TRACE codes. First, the Maanshan TRACE model was developed to analyze the LBLOCA. The analysis results were benchmarked and compared with Maanshan FSAR data. The input file of FRAPTRAN mainly composes of three parts to define the transient problems: 1. Fuel rod geometry; 2. Power history; 3. Coolant boundary conditions. The geometry data of the fuel rod and the results from TRACE analysis (e.g. fuel rod power, coolant pressure, heat transfer coefficient) were inputted into FRAPTRAN to analyze the reliability of fuel rod. FRAPTRAN can obtain the detail mechanical property of fuel rod (e.g. cladding temperature, hoop stress/strain, gap pressure, and oxide thickness of cladding). In order to do uncertainty analysis, it must to identify and select the important parameters in the specific transient. DAKOTA code[4] is used to generate random variable and to evaluate response data generated by FRAPTRAN/SNAP. The sampling method in DAKOTA could choose the Monte-Carlo or Latin Hypercube method. Because the required minimum number of FRAPTRAN runs is dependent of the values of confidence level and probability, Wilks' formula[5] was employed to determinate the minimum number of runs. Since the value of Peak Cladding Temperature (PCT) is the safety criterion to ensure the integrity of fuel assemblies for LOCAs, the minimum number of 59 was used to generate the maximum bound of PCT which achieve 95/95 criterion. All FRAPTRAN runs were defined and executed through SNAP job streams[6-7]. The data interactions and communications between FRAPTRAN and DAKOTA were controlled by SNAP.

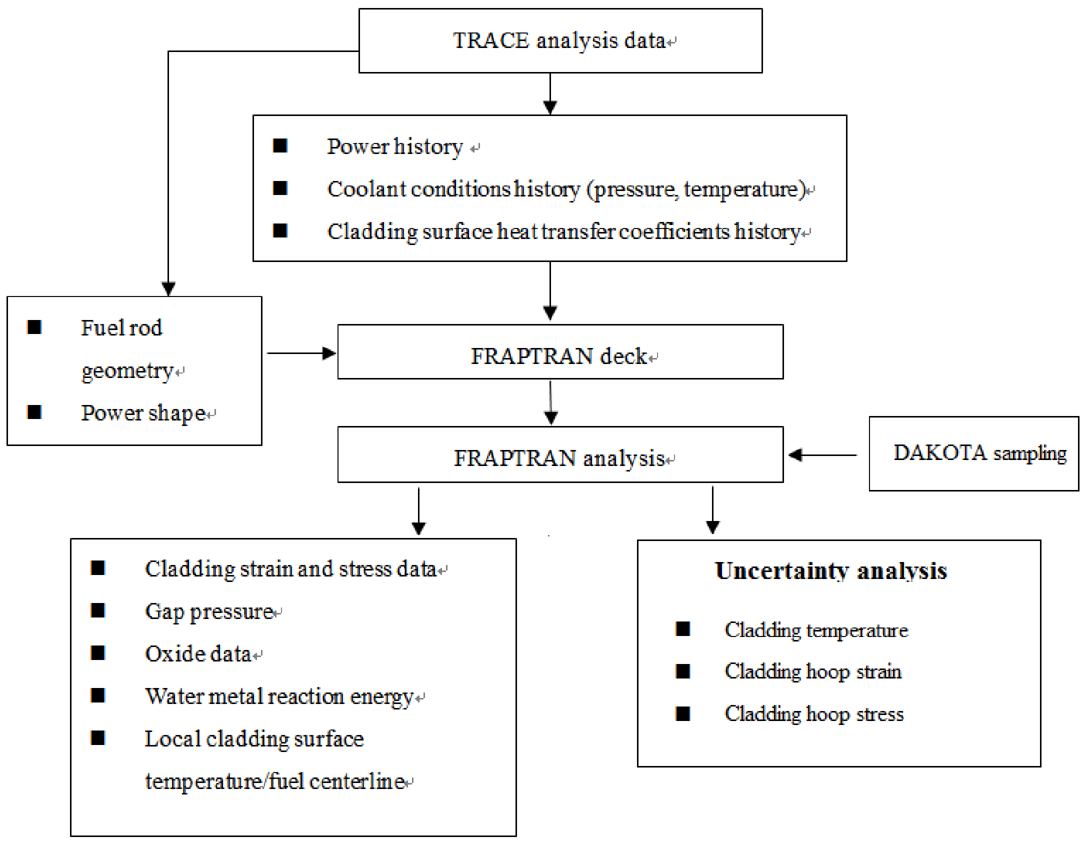

Figure 1 The Flowchart of Combining TRACE /FRAPTRAN/DAKOTA Codes

2.2 Analysis Tools

TRACE (TRAC/RELAP Advanced Computational Engine) is an advanced and best-estimate reactor systems code for analyzing thermal hydraulic behaviors in light water reactors[8]. TRACE consolidates the capabilities of the four codes, TRAC-P, TRAC-B, RELAP 5 and RAMONA, into one modernized code. One of the most important features of TRACE is its capability 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 for LOCA.

Fuel Rod Analysis Program Transient (FRAPTRAN)[9-10] is a FORTRAN language computer code which was developed by Pacific Northwest National Laboratory, PNNL. The main purpose of this code is to calculate the response of a single fuel rod transient performance in Light Water Reactors (LWR) during operational transients or hypothetical accidents, such as Reactivity Accidents (RIA) or LOCA, up to burnup level of 62 GWd/MTU. FRAPTRAN is also a companion code to the FRAPCON-3 which was developed to calculate steady-state high burnup level response of a single fuel rod. FRAPTRAN calculates the fuel and cladding temperatures, cladding strain and stress, and plenum gas pressure at different time for given power and coolant conditions.

FRACAS-I model in FRAPTRAN is used to calculate the mechanical responses of the fuel rod and cladding. The failure models in FRAPTRAN apply to LOCA events where either a deformation due to gas overpressure or the relatively high cladding temperature (>700 K). After the cladding effective plastic strain is calculated by FRACAS-I, this value is compared with the instability strain given by MATPRO. If the cladding effective plastic strain is greater than the instability strain, the ballooning model, BALON2, is used to calculate the localized, non-uniform strain of cladding. The BALON2 model has two criteria to predict failure in the ballooning node: one is when the cladding hoop stress exceeds an empirical limit; the other is when predicted cladding permanent hoop strain exceeds the FRAPTRAN strain limit.

In uncertainty analysis, the DAKOTA (Design Analysis Kit for Optimization and Terascale Application) code is developed by the Sandia National Laboratories. DAKOTA is described as a multilevel parallel object-oriented framework for design optimization, parameter estimation, uncertainty quantification, and sensitivity analysis.

SNAP is a graphical user interface program that provides users an easy way for FRAPTRAN to input the parameters, execute and create the output result file automatically in the route that users indicate right after the end of the calculation.

3. TRACE ANALYSIS RESULTS

There were three main steps in this research. The first step was the LBLOCA analysis for Maanshan NPP by TRACE code. The analysis results were benchmarked and compared with Maanshan FSAR data. In second step, it used FRAPTRAN to calculate the response of a single fuel rod transient behaviour during LBLOCA. After all, uncertainty analysis was considered in this study.

3.1 Maanshan TRACE Model

Figure 2 shows the TRACE model of Maanshan NPP. It is a three-loop model, and each loop has a feedwater control system. The main structure of this model includes the pressure vessel, pressurizer, steam generators, steam piping at the secondary side, the steam dump system, accumulators, and safety injection of ECCS. The pressure vessels are cylindrical, and its divisions are as shown in Figure 2 . It is divided into 12 levels in the axial direction, two rings in the radial direction (internal and external rings) and six equal azimuthal sectors in the “θ” direction. The control rod conduit connects the 12th and 7th layers of the vessel from end to end. The fuel region is between the third and sixth layers, and heat conductors were added onto these structures to simulate the reactor core.

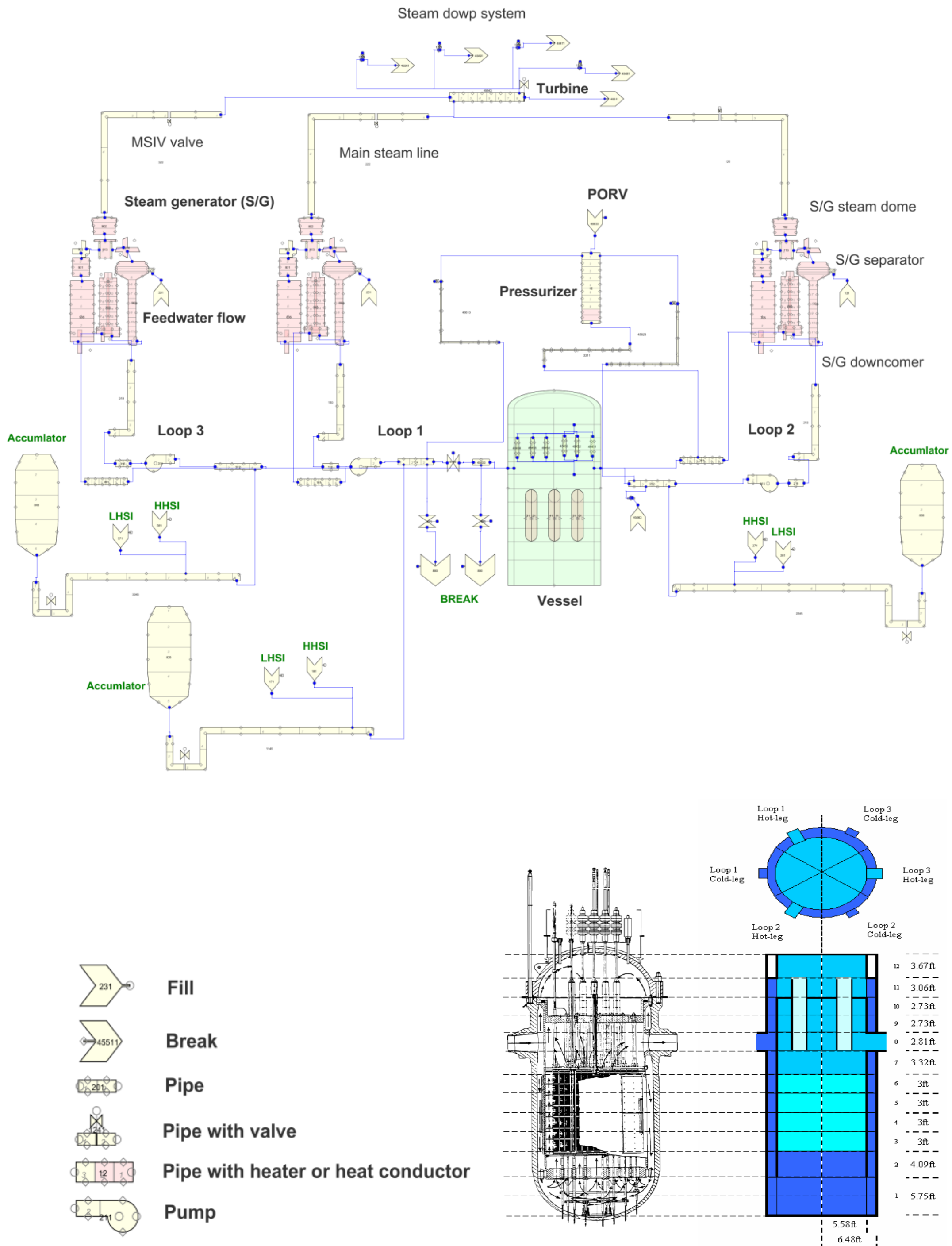

Figure 2 The LBLOCA TRACE Model of Maanshan NPP

3.2 LBLOCA Analysis Results

Figure 3 plots the power curve that calculated from TRACE in the case of LBLOCA, and then compares with the FSAR data. In TRACE, the core power can be calculated using the built-in point kinetics model, and the power calculated includes decay heat. It displays that the power curve of TRACE is almost the same as those of FSAR data. Figure 4 compares the pressures of the vessel and suggests that the pressure calculated by TRACE approximately follows the trend of the FSAR data. Figure 5 compares the break mass flow rate of cold-leg pipe. It reveals that break mass flow rate predicted by TRACE agrees closely with the results of the FSAR data. Figure 6 shows the comparisons of accumulator mass flow rate of intact loops between TRACE model and FSAR data. Figure 7 compares the core inlet flow rate, revealing that the flow rate calculated by TRACE is in agreement with the FSAR data except for the period between 6 and 18 sec. It reveals that the flow rate calculated by TRACE is slightly lower between 6 and 18 sec. Figure 8 plots the results for core outlet flow rate. The difference results of core outlet flow before 6 sec are consideration of the nature flow in TRACE. Analytical results indicate that the Maanshan TRACE model predicts the behaviours of important plant parameters in consistent trends with the FSAR data.

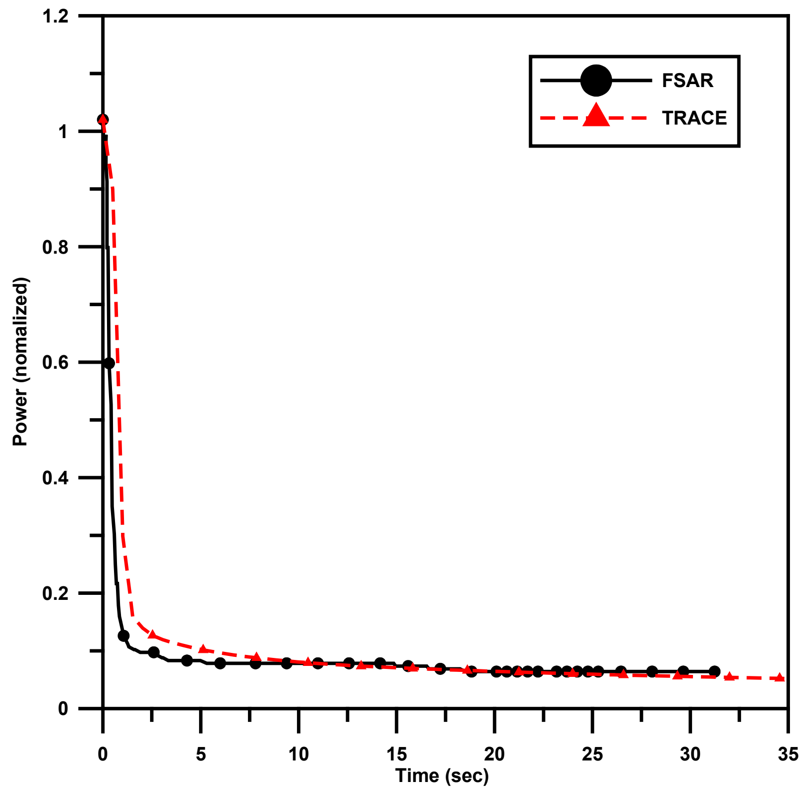

Figure 3 The Results of Power

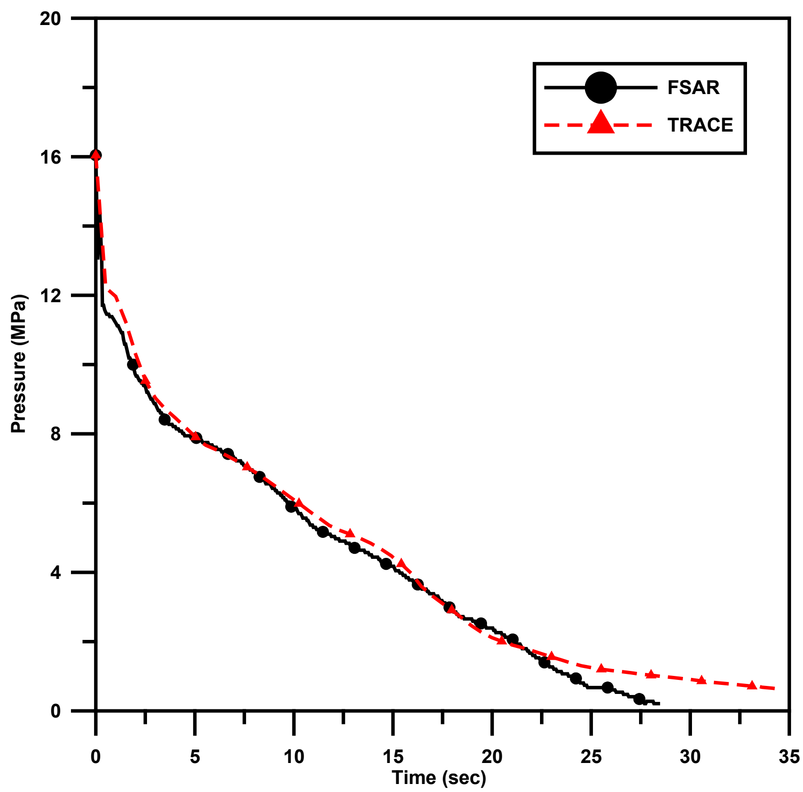

Figure 4 The Results of Vessel Pressure

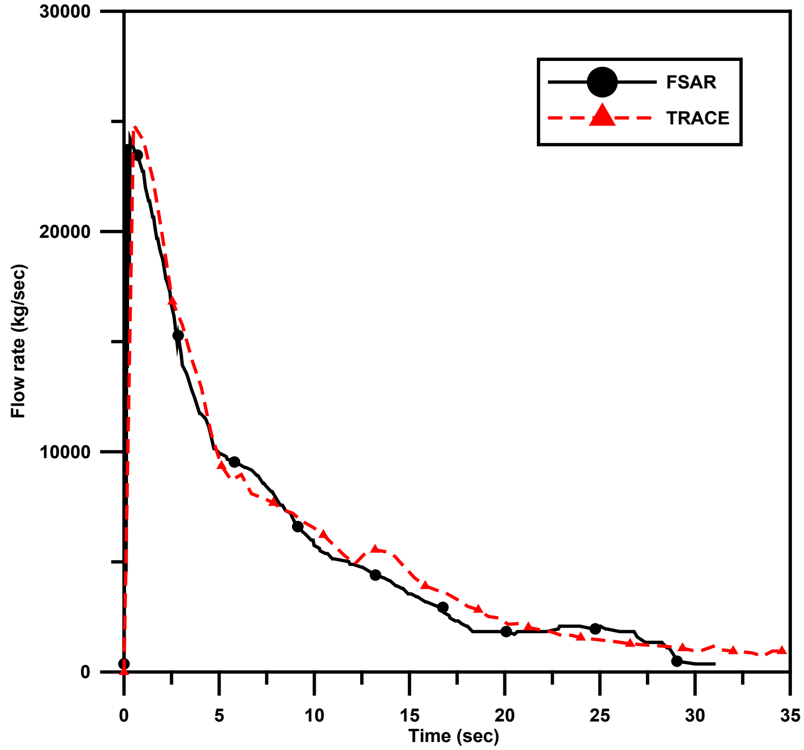

Figure 5 The Results of Break Mass Flow Rate

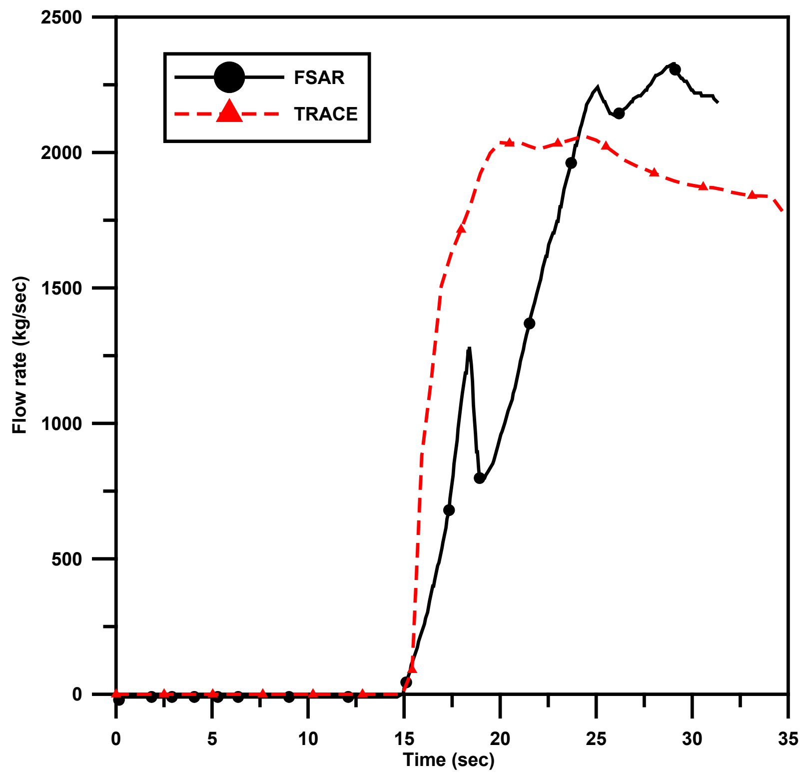

Figure 6 The Results of Accumulator Flow Rate

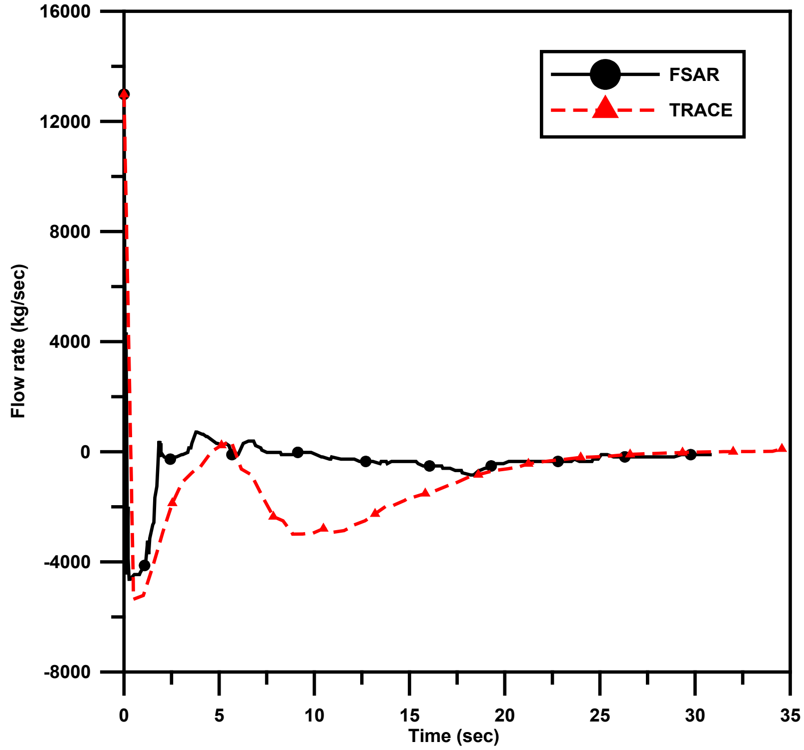

Figure 7 The Results of Core Inlet Flow

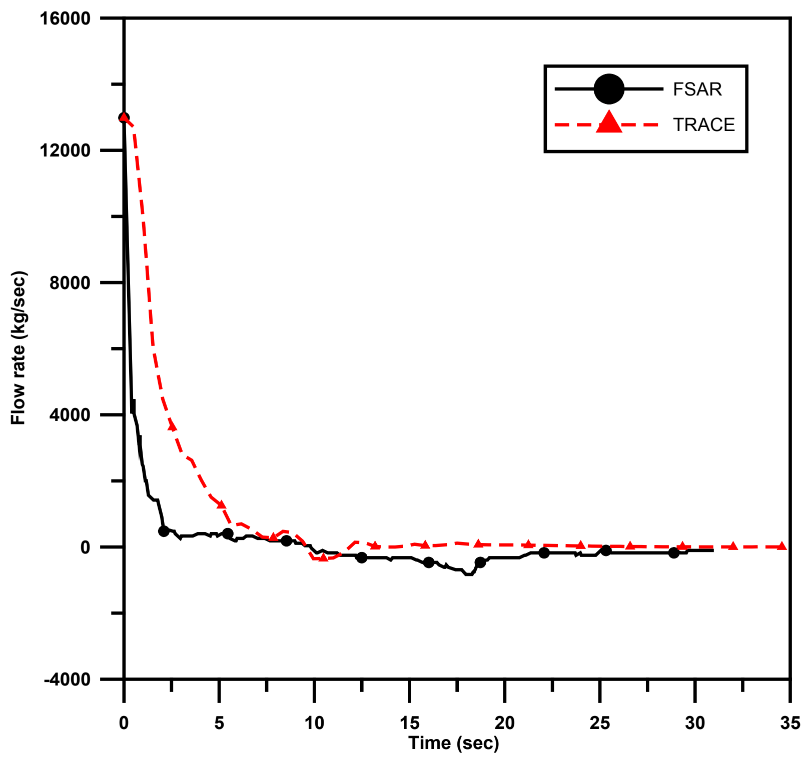

Figure 8 The Results of Core Outlet Flow

4. FRAPTRAN ANALYSIS RESULTS

4.1 FRAPTRAN Parameter Description

The fuel rod design parameters were determined according to the fuel rod that Maanshan NPP is using at present. A new rod was assumed, and the rod design parameters are listed in Table 1. (This assumption may be not conservative, it will consider the burn-up of the fuel rod in the future study.) Figure 9 illustrates the schematic of fuel rod in FRAPTRAN. The axial fuel length from bottom to top was divided into 12 nodes from bottom to top and the fuel radial direction was divided into 17 nodes, including 15 nodes in the pellet and 2 nodes in the cladding.

The vessel structure model of Maanshan NPP in TRACE is shown in Figure 10. The vessel was divided into 2 rings in the radial direction, 6 parts in the azimuthal direction and 12 levels along the axial direction from bottom to top. Six heat structures were set at the 3-6 levels which located separately in the 6 different azimuthal zones of the ring 1 as heat sources in the vessel. Thus, the results from these 6 heat structures offered the essential data and heat transfer coefficient for input file. In what follows, the transient results of the fuel rods from these different zones (referred as T01-T06) will be discussed.

According to the setting in TRACE, the reactor scramed at 0.5 s while the reactor pressure reached 12.8 MPa. The Figure 11 and Figure 12 show the power and boundary condition settings of FRAPTRAN which are offered by TRACE results during the accident.

In Figure 11, we can read from power history that the power drops sharply after scram during the accident and the axial power ratio distribution is similar to a consine shape.

“Heat” option was chosen as the boundary condition setting in FRAPTRAN. The option requires coolant pressure, temperature and heat transfer coefficient which varied with time to describe the boundary condition during accident. The coolant temperature which is shown in Figure 12 drops gradually and then maintains at the saturation temperature 40 MPaG respectively, which is below the 10.342 MPaG limit.

Table 1 Fuel Rod Design Parameters

Fuel rod parameters	**Values**
Fuel active length	3.6576 m
Fuel rod OD	9.14 x 10^{-3} m
Cladding type	ZIRLO
Cladding thickness	0.5715 x 10^{-3} m
Filling gas	He, 1.551 MPa

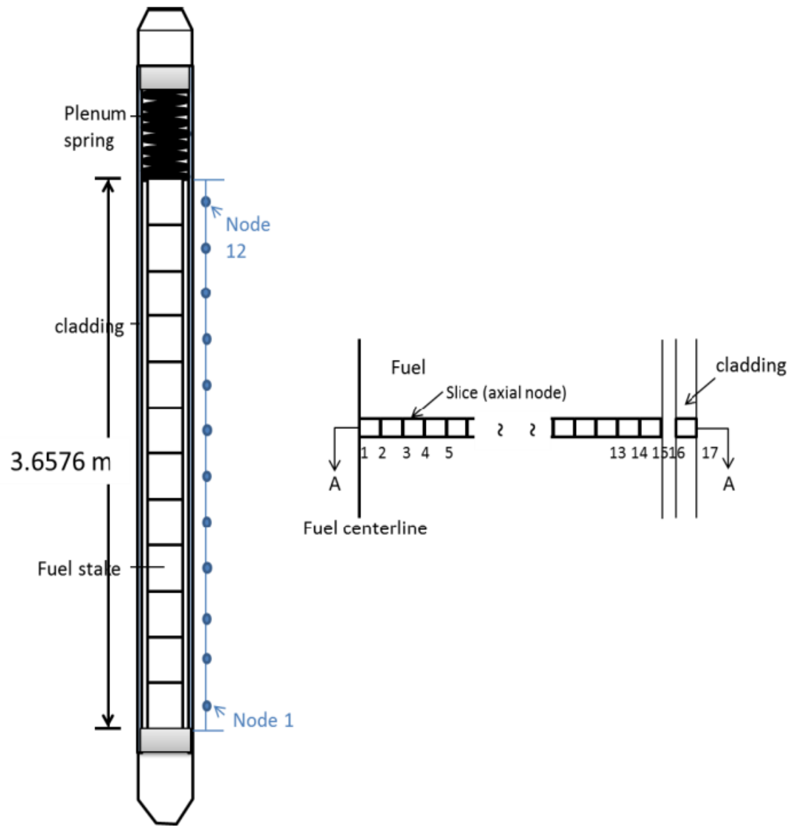

Figure 9 The Schematic of Fuel Rod Geometry In FRAPTRAN

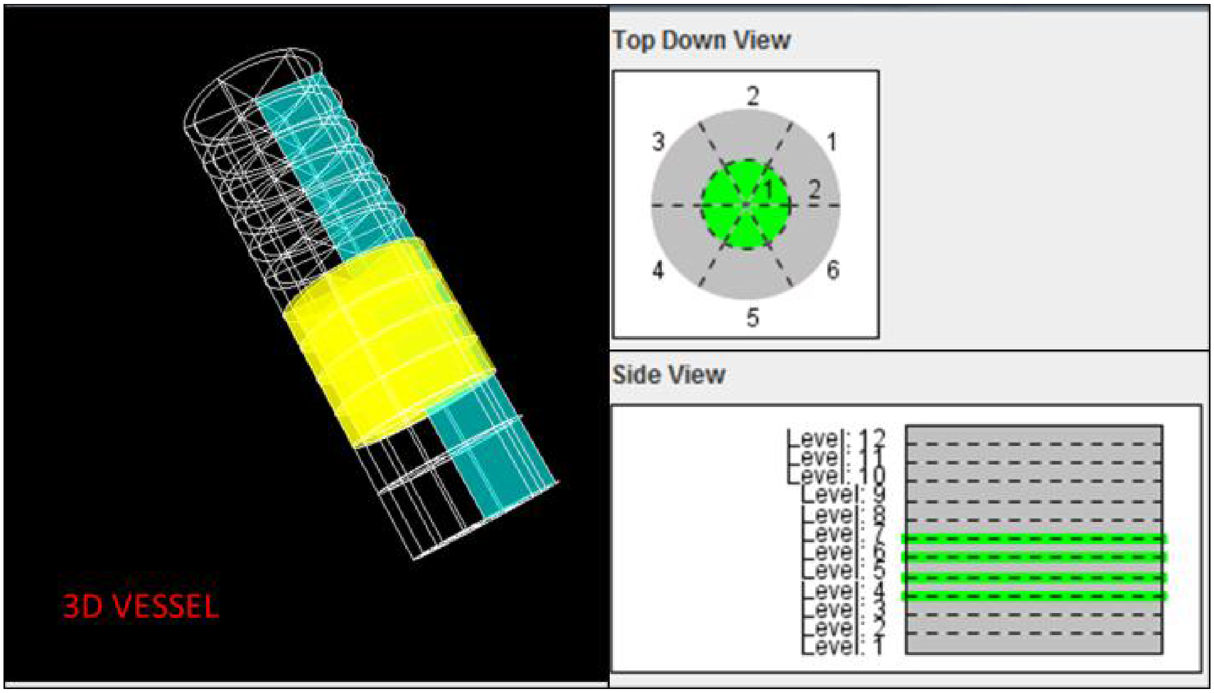

Figure 10 The Vessel Structure Model in TRACE

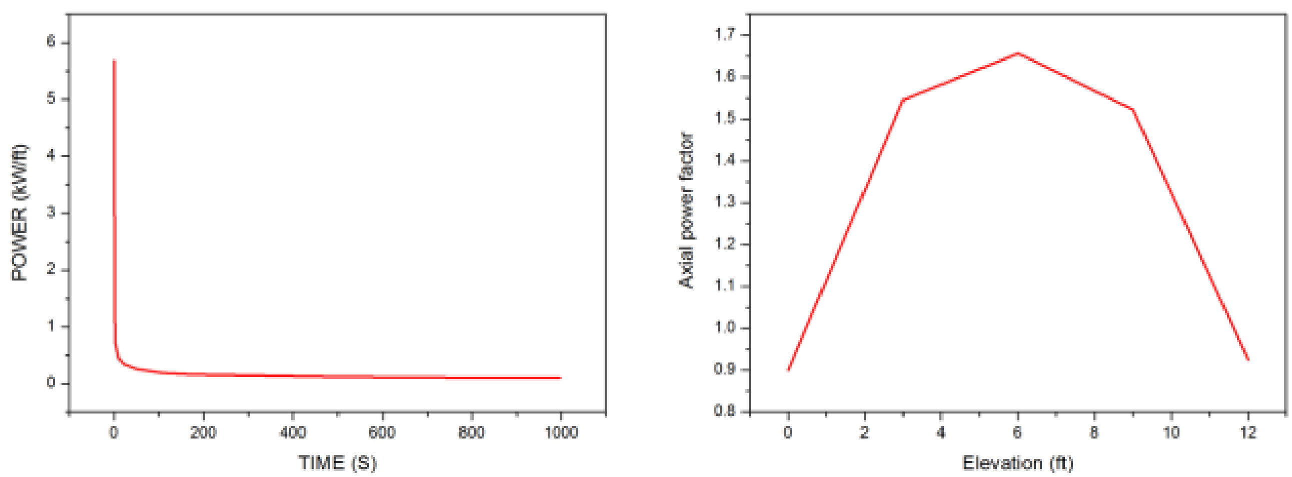

Figure 11 The Power History after Scram from TRACE (Left) and the Axial Power Shape of the Fuel Rod (Right)

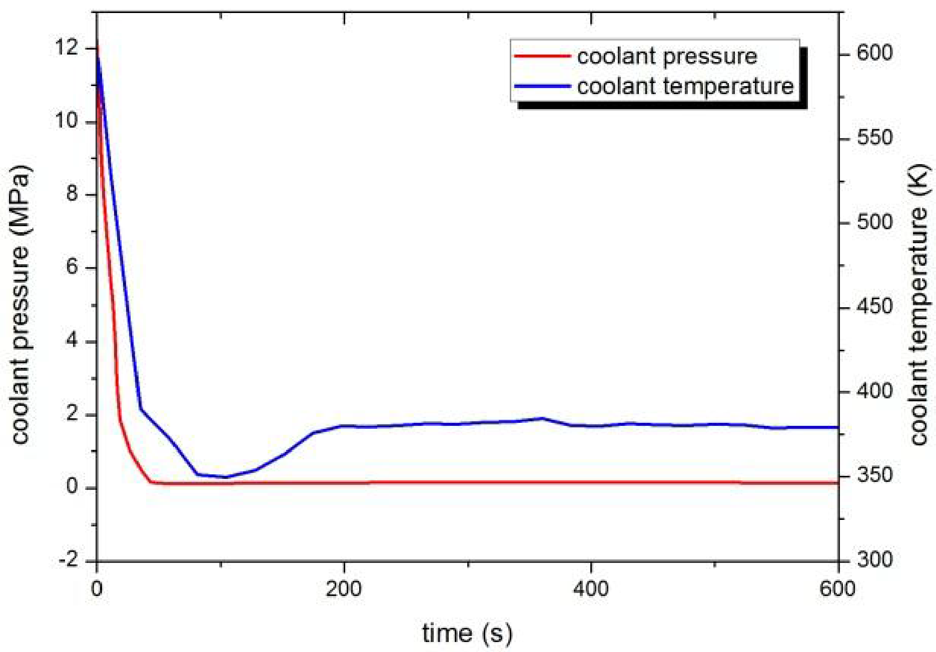

Figure 12 The Coolant Pressure and Temperature History at T01 Zone from TRACE

4.2 Analysis Results

Figure 13 shows the simulation results of gap width between the fuel pellet and cladding under the open gap situation without pellet cladding interaction mechanisms. The plenum pressure variation at T01 zone is shown in Figure 14. The gas gap pressures at different nodes are not shown in this figure since the results at different elevations are all the same. The plenum pressure drops from 4.31 MPa to 2.2 MPa in the end of the transient, and it appears that there is no failure of the fuel rod. According to the above results of the gap width and the gas gap pressure variations, one can conclude that the ballooning phenomenon may not occur in this case. Furthermore, the hoop stress simulation results of T03 zone are shown in Figure 15. The hoop stress results of different nodes are nearly the same, and the hoop stress in each zone (T01-T06) appears about 18 MPa, which is well below the limitation of material properties. Therefore, it is ensured that the load to the cladding is still within the safety range.

As mentioned in the introduction section, the safety criteria of 10CFR50.46 for a LOCA scenario in a nuclear reactor, the ECCS must be designed and activated so that PCT should not exceed 1477.6 K (2200˚F). In what follows, the results of fuel centerline temperature and the cladding temperature during LBLOCA are discussed.

Figure 16 shows the fuel centerline temperatures of nodes 1, 4, 7, 10, 12. The fuel centerline temperature is around 10 K higher than the outside cladding temperature. It's obvious that the peak cladding temperatures and the fuel centerline temperature were all below the criteria requirement. According to the safety criteria, the oxide thickness should be less than 17% of the total the cladding thickness, which is 0.57531 mm in this case. The simulation results of the oxide thickness were 0.006 mm for the present case. Even considering the variation of the cladding thickness during transient, the Effective Cladding Reacted (ECR) is only about 1%.

The cladding temperature results in node 1, 4, 7, 10, 11 at T01 to T06 zones are shown in Figure 17. According to the present FRAPTRAN simulations for the fuel rods at six zones in the vessel, the highest temperature spots of the fuel rod were found at node 4 (1.067 m) in T01, T04, T05 and T06 zones and node 7 (1.981 m) in T02 and T03 zones. The peak cladding temperature of the six zones is around 800 K and then drop down to around 400 K at the end of the transient. In addition, it is found that all of the results show the similar trends and orders of response values to those in literature (Ref. 8, 9), which suggests that important physical phenomenon was properly simulated in the present case. The present results well meet the safety criteria required for cladding temperature (<1477.6 K).

According to the safety criteria, the oxide thickness should be less than 17% of the total the cladding thickness, which is 0.57531 mm in this case. The simulation results of the oxide thickness in six zones were all the same as 0.006 mm for the present case. Even considering the variation of the cladding thickness during transient, the ECR (effective cladding reacted) is only about 1%. The value is still within the safety operation requirement as mentioned above. Note that the zero-burnup rod was considered in this study, and the resultant cladding temperature did not reach high enough temperature for violent metal-water reactions.

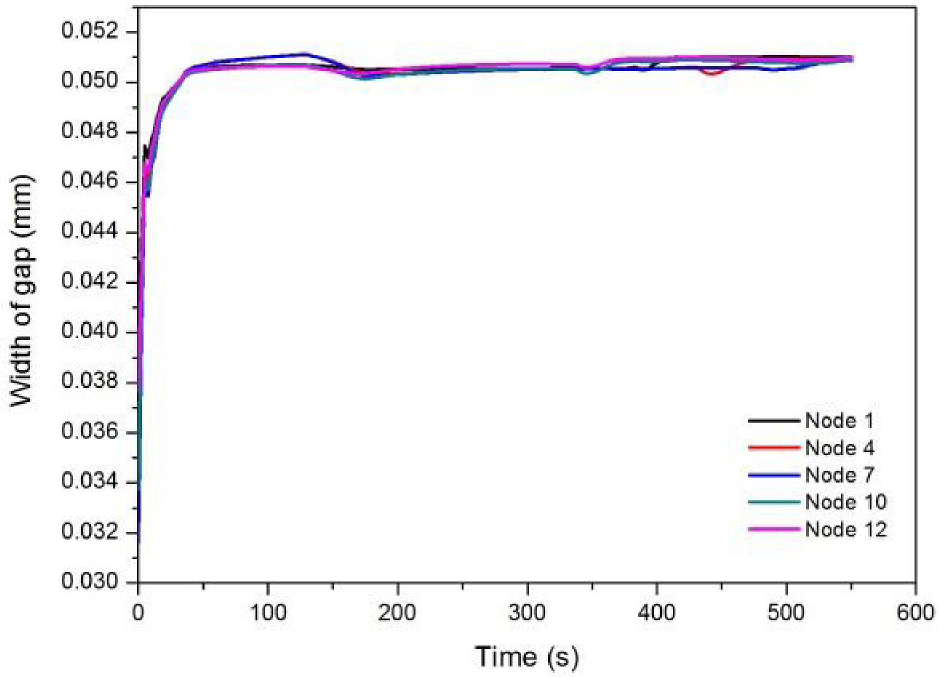

Figure 13 The Variations of Gap Width in the Rod during Transient at T03 Zone

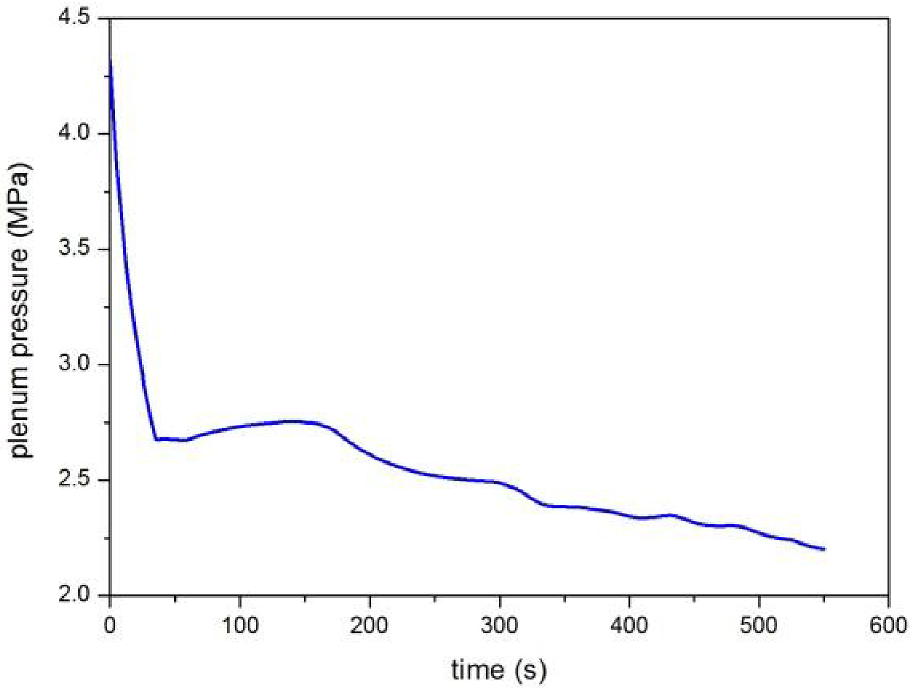

Figure 14 The Plenum Pressure Variation during Transient at T01 Zone

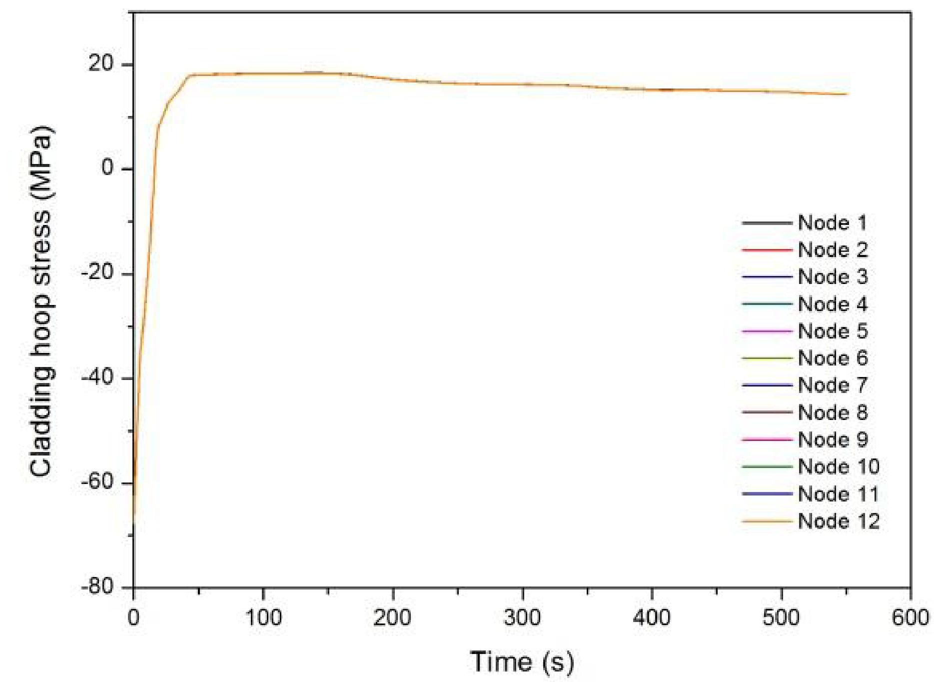

Figure 15 The Hoop Stress Variations during Transient at T03 Zone

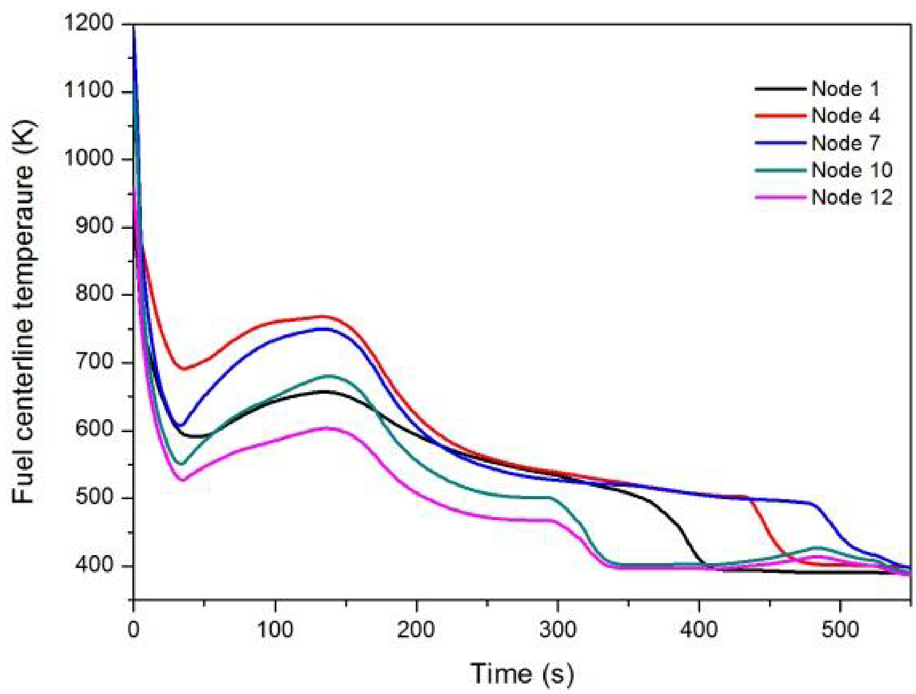

Figure 16 The Fuel Centerline Temperatures at T01 Zone

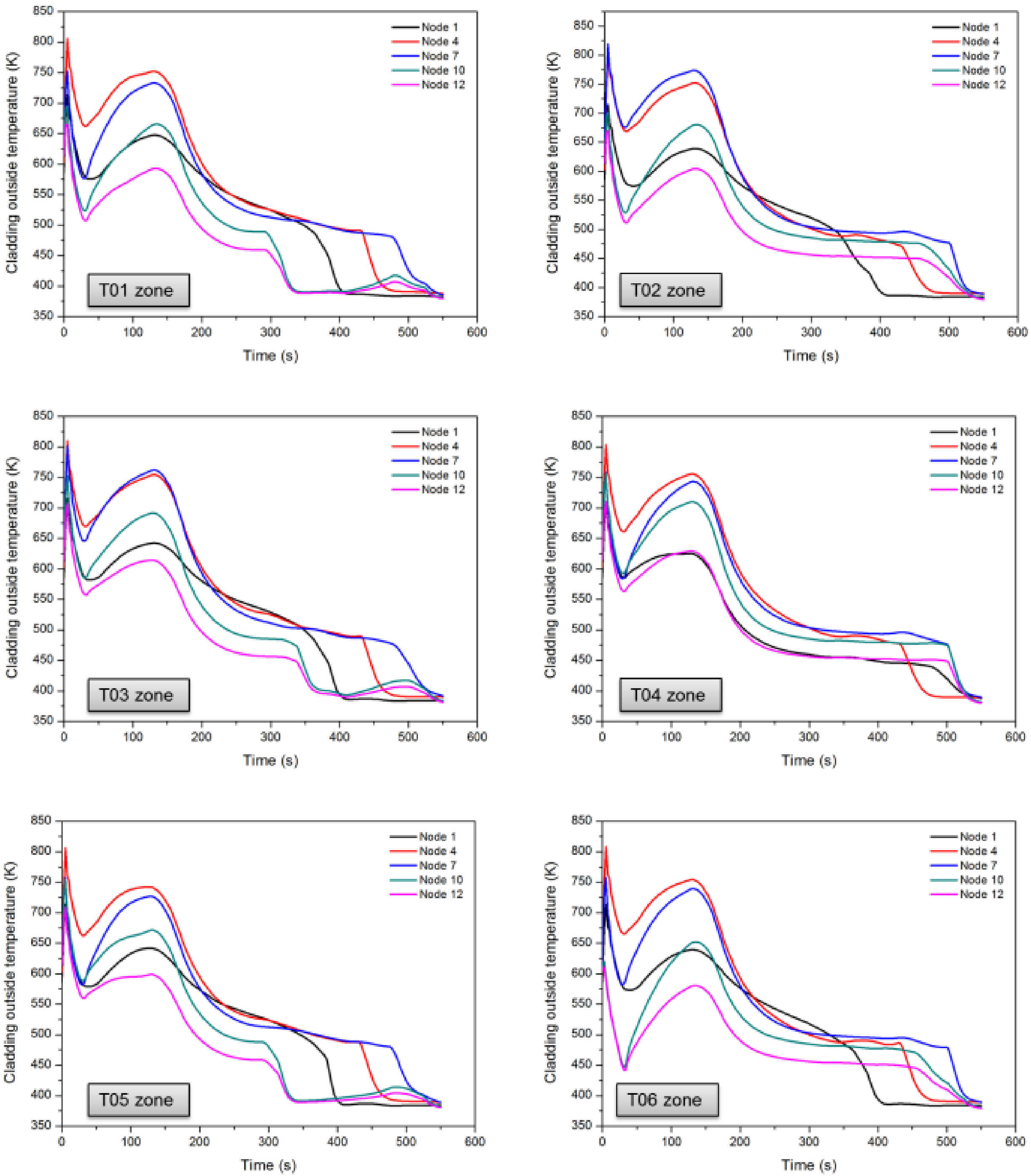

Figure 17 The Cladding Temperatures at T01-06 Zones

4.3 SNAP Animation

SNAP program provides the function of the animation presentation for FRAPTRAN. In the animation model, the performance of the fuel rod during transient can be presented in axial cross sectional view with the temperatures at different nodes in axial and radial directions in each time step as well as making X-Y plots of any parameters of interests. The radius of each radial node can be calculated and displayed, so that one can see noticeable changes when ballooning phenomenon occurs.

Figure 18 illustrates the animation results at time of 100 s, 300 s, 500 s (from left to right) for T01 zone. The histories of fuel centerline temperatures are also shown below each fuel rod. It should be mentioned that node 4 shows the highest temperature at 100 s. The variations of temperature also indicate obviously that the cooling process started from the fuel top first at 300 s, and then from the bottom at 500s during LBLOCA.

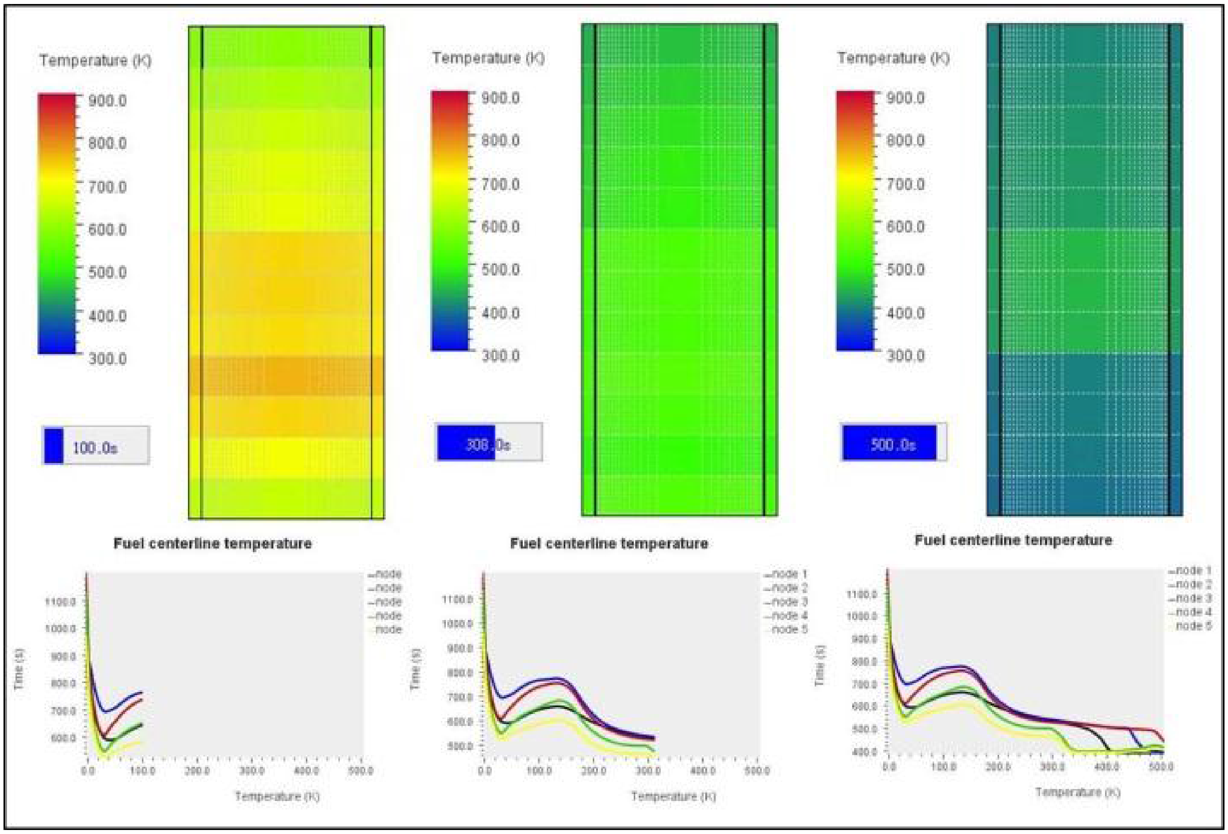

Figure 18   Animation Presentation in SNAP Interface (T01 Zone)

5. UNCERTAINTY ANALYSIS

In uncertainty analysis, the several parameters of fuel rod, such as fabrication and boundary conditions, were quantized and sampled by the DAKOTA uncertainty code. Table 2 lists the 5 importance parameters (i.e., Empty reactor time, flooding start time, Gap thickness, Fill gas temp, and pellet diameter) taken into account in this uncertainty analysis, which are defined as the SNAP user-defined numeric variables and linked with uncertainty configuration to generate input files. By coupling with DAKOTA, the important parameters with uncertainties were generated randomly based on specified PDFs. In particular, the statistical theory predicts that 59 calculations are required to simultaneously bound the 95th percentile of one parameters with a 95-percent confidence level.

Figure 19 displays the 59 cladding temperatures as a function of time. The uncertainty band is formed by the 59 calculations. The minimum and maximum PCT are 842K and 892K respectively. The uncertainty analysis shows that the PCT were all below the criteria requirement. Figure 20 and Figure 21 show the hoop strain and stress simulation results. The maximum hoop strain and hoop stress are 0.006 and 37 MPa, which are all well below the limitation of material properties. Therefore, it is ensured that the load to the cladding is still within the safety range.

Table 2 The 5 Importance Parameters in Uncertainty Analysis

	Mean	Lower bound	Upper bound	Distribution
Reactor conditions				
Empty reactor time (s)	13.292	-1	+1	uniform
Flooding start time (s)	164.27	-5	+5	uniform
Rod design				
Gap thickness(m)	7.5E-5	-2%	+2%	uniform
Fill gas temp (K)	300	-2%	+2%	uniform
Fuel pellet				
pellet diameter (m)	0.00784	-2%	+2%	uniform

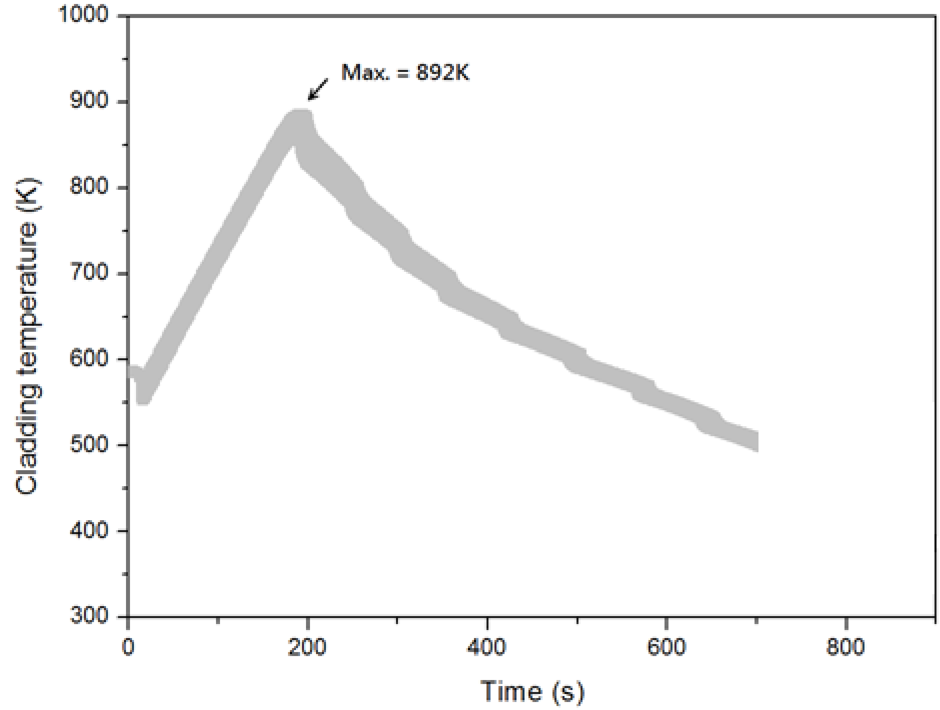

Figure 19 The 59 Cladding Temperatures as A Function of Time

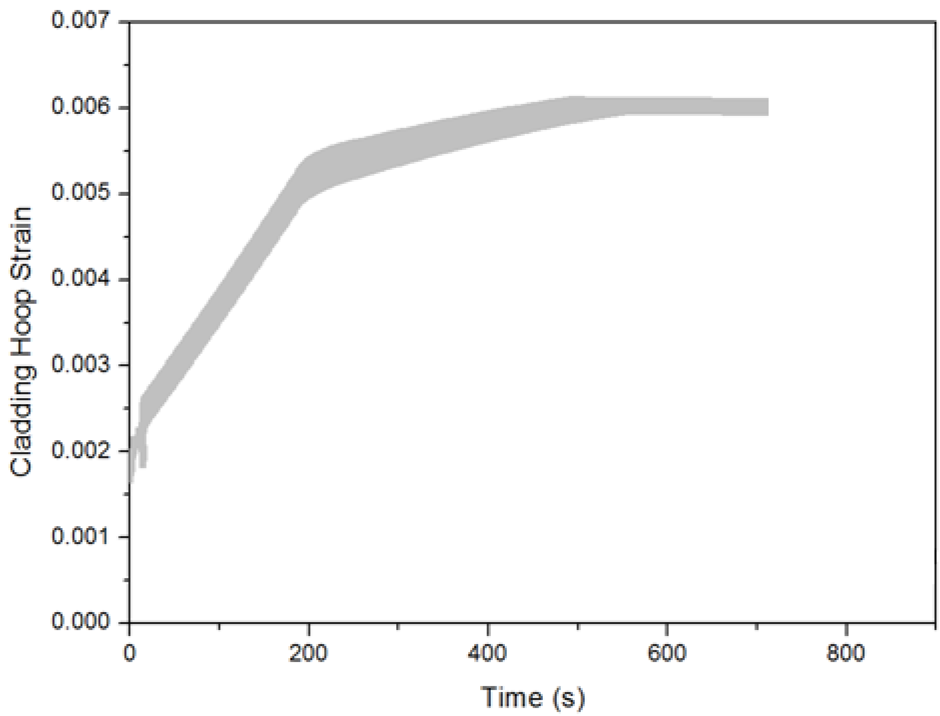

Figure 20 The 59 Cladding Hoop Strains as A Function of Time

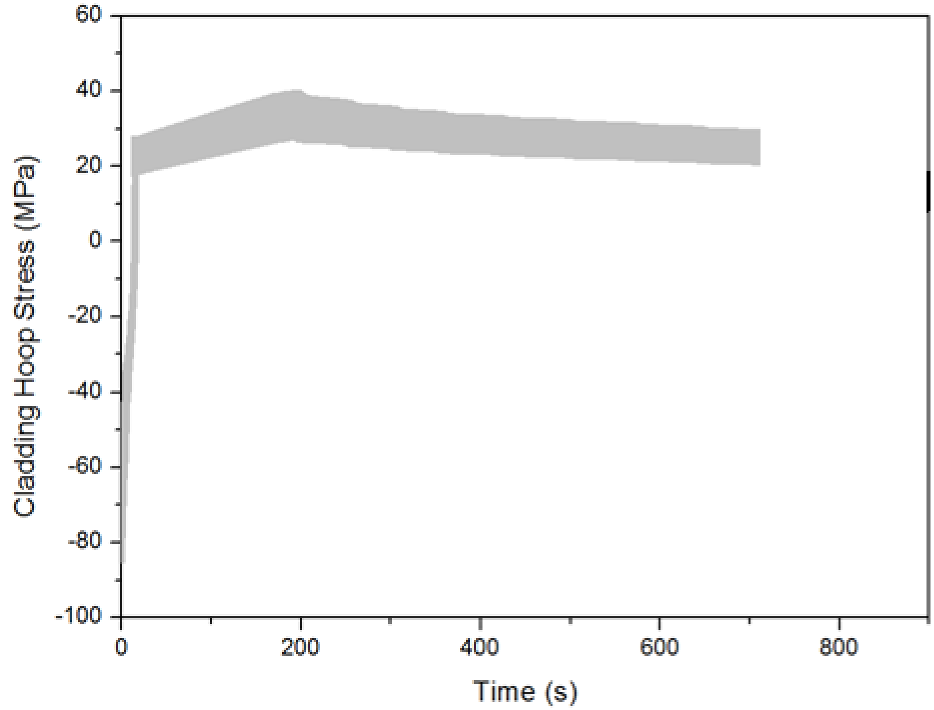

Figure 21 The 59 Cladding Hoop Stresses as A Function of Time

6. CONCLUSIONS

In this paper, the TRACE/FRAPTRAN code under SNAP program was used to analyze the behaviour of the fuel rod during LBLOCA transient at Maanshan NPP. Important results can be summarized as below:

1. In first task, the thermal-hydraulic analytical results indicate that the Maanshan TRACE model predicts the behaviors of important plant parameters in consistent trends with the FSAR data.
2. The FRAPTRAN code and TRACE results have been successfully combined to analyze the fuel rod transient behaviors in the LBLOCA scenario.
3. In FRAPTRAN analysis, the plenum pressure drops from 4.31 MPa to 2.2 MPa in the end of the transient, and it appears that there is no failure of the fuel rod.
4. The hoop stress results of different nodes are nearly the same, and the hoop stress appears about 18 MPa, which is well below the limitation of material properties. Therefore, it is ensured that the load to the cladding is still within the safety range.
5. The oxide thickness calculation is less than 17% of the total the cladding thickness. Based on the calculation results of plenum pressure and the gap thickness, there was no cladding ballooning or failure in the present case.
6. According to the present FRAPTRAN simulations for the fuel rods arranged in six zones in the vessel, the highest (centerline) temperature spots of the fuel rods were found at node 4 of T01, T04, T05 and T06 zones and node 7 of T02 and T03 zones. The peak cladding temperatures of each zone were found at the same nodes with that of the centerline temperatures. The cladding temperatures and the fuel centerline temperatures were all below the criteria requirement.
7. Considering uncertainty analysis, the uncertainty band is formed by the 59 calculations. The minimum and maximum PCT are 842K and 892K respectively. The uncertainty analysis shows that the PCTs were all below the criteria requirement. The maximum hoop stress was about 37 MPa and the maximum hoop strain was 0.006 for the fuel rod simulation, which are all within the safety operation range.
8. The animation function in SNAP program provides a clear and easy way to monitor the fuel performance during transient. One can identify if cladding ballooning occurs through color change in different areas and radius change in radial direction.

In conclusion, the present study confirms that this analysis method, the FRAPTRAN code combined with TRACE results, is an appropriate approach to predict the fuel integrity under LBLOCA with operational ECCS.

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11. ABSTRACT (200 words or less)

The Maanshan NPP operated by Taiwan Power Company (TPC) is the only Westinghouse-PWR in Taiwan. The rated core thermal power is 2775 MW. The reactor coolant system has three loops, each of which includes a reactor coolant pump and a steam generator. The pressurizer is connected to the hot-leg piping in loop 2. The main components of Maanshan TRACE model include the pressure vessel, pressurizer, steam generators, steam piping in the secondary side (including four sets of steam dump and vent valves), the steam dump system, accumulators, and safety injection of emergency core cooling system (ECCS). The pressure vessel is divided into 12 levels in the axial direction, two rings in the radial direction (internal and external rings) and six equal azimuthally sectors in the "θ" direction. The control rod conduit connects the 12th and 7th layers of the vessel from end to end. The fuel region is between the third and sixth layers, and heat conductors are added onto these structures to simulate the reactor core. In this study, the FRAPTRAN and TRACE code were used to evaluate the fuel rod transient behavior during a postulated LBLOCA in Maanshan (3-loops PWR) nuclear power plant. There were three main steps in this research. The first step was the LBLOCA analysis for Maanshan NPP by TRACE code. The analysis results were benchmarked and compared with Maanshan FSAR data. In second step, it used FRAPTRAN to calculate the response of a single fuel rod transient behavior during LBLOCA. After all, uncertainty analysis was considered in this study. The several parameters of fuel rod, such as fabrication and boundary conditions, were quantized and sampled by the DAKOTA uncertainty code.

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Best Estimate plus Uncertainty (BEPU)
Main Steam Isolation Valve (MSIV)
Maanshan Nuclear Power Plant (NPP)

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