International Journal of Engineering, Mathematical and Physical Sciences
ISSN: 2517-9934
Vol:8, No:1, 2014

The Analysis of TRACE/FRAPTRAN in the Fuel Rods of Maanshan PWR for LBLOCA

J. R. Wang, W. Y. Li, H. T. Lin, J. H. Yang, C. Shih, S. W. Chen

***Abstract*—**Fuel rod analysis program transient (FRAPTRAN) code was used to study the fuel rod performance during a postulated large break loss of coolant accident (LBLOCA) in Maanshan nuclear power plant (NPP). Previous transient results from thermal hydraulic code, TRACE, with the same LBLOCA scenario, were used as input boundary conditions for FRAPTRAN. The simulation results showed that the peak cladding temperatures and the fuel centerline temperatures were all below the 10CFR50.46 LOCA criteria. In addition, the maximum hoop stress was 18 MPa and the oxide thickness was 0.003mm for the present simulation cases, which are all within the safety operation ranges. The present study confirms that this analysis method, the FRAPTRAN code combined with TRACE, is an appropriate approach to predict the fuel integrity under LBLOCA with operational ECCS.

***Keywords*—**FRAPTRAN, TRACE, LOCA, PWR.

I. Introduction

THE Maanshan NPP operated by Taiwan Power Company has two Westinghouse PWR units. The rated core thermal power of each unit is 2775 MWt. 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 analysis of LBLOCA scenario for the Maanshan NPP with TRACE code has been published previously [1], [2]. The 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.

LOCAis one of the important Design Basis Accidents (DBAs) in light water reactors, and the 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 Emergency Core Cooling System (ECCS) requirements. LBLOCA scenario is considered as the worst one. Although it is considered very unlikely to occur, the safety systems must be designed to secure adequate cooling of the reactor core. Referring to 10CFR50.46, the current LOCA criteria are briefly listed below [3]:

1. The peak clad temperature shall not exceed 1477.6K.
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 [4], [5]. In this research, the previous TRACE results of Maanshan NPPLBLOCA 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.

TRACE has been developed by U.S.NRC for NPP safety analysis. In the future, TRACE is expected to replace NRC's present four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) as the main code used in thermal hydraulic safety analysis [2].

FRAPTRAN is a FORTRAN language computer code which was developed by Pacific Northwest National Laboratory (PNNL) [6], [7]. 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 [6].

SNAP is a graphical user interface program that provides users an easy way for TRACE and FRAPTRAN to input the

J. R. Wang is with the Institute of Nuclear Energy Research, Atomic Energy Council, and the Institute of Nuclear Engineering and Science, National Tsing-HuaUniversity,R.O.C., Taiwan (e-mail: jrwang@iner.gov.tw).
W.Y. Li and J.H. Yangare with the Institute of Nuclear Engineering and Science, National Tsing-Hua University, R.O.C., Taiwan (corresponding author e-mail: fermmy@hotmail.com, junghua1984@gmail.com).
H.T. Lin is with the Institute of Nuclear Energy Research, Atomic Energy Council, R.O.C., Taiwan (e-mail: htlin@iner.gov.tw).
C. Shih and S.W. Chenare with the Institute of Nuclear Engineering and Science, National Tsing-Hua University, and Nuclear Information Center, R.O.C., Taiwan (e-mail:ckshih@ess.nthu.edu.tw).

parameters, execute and create the output result file automatically after the end of the calculation.

II. TRACE/FRAPTRAN Modeling of Maanshan PWR

Fig. 1 shows the TRACE model of Maanshan NPP. Maanshan NPP TRACE model contains 69 hydraulic components, 380 control blocks, 34 heat structures and 2 power components. Main components including one 3-D vessel, three RCS loops, one pressurizer, three steam generators and basic plant control systems such as 3-element feedwater control, pressurizer spray, pressurizer level and heater control, and steam dump control. The 3-D vessel component contains 2 radial rings, 6 azimuthal sectors and 12 axial levels. The outer radial ring represents downcomer region and the reactor core is placed in the inner radial ring from axial level 3 to axial level 6. Six control rod guide tubes are connected above the core region. Nuclear fuels are modeled by 6 heat structures each represents 6908 average fuel rods that uniformly placed in 6 azimuthal sectors. Each RCS loop contains hot leg piping, steam generator U-tube, crossover piping, reactor coolant pump, cold leg piping, accumulator tank and accumulator check valve. Pressurizer and pressurizer surge line are connected on RCS loop 2.

The usage of FRAPTRAN under SNAP interface combined with TRACE analysis results to analyze the fuel rod performance for LBLOCA transient is the main strategy in this paper. Fig. 2 shows the flowchart of combining FRAPTRAN and TRACE codes. 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 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 I. Fig. 3 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. By doing so, one can obtain the detail temperature distributions inside the fuel rod during any transients. The vessel structure model of Maanshan NPP in TRACE is shown in Fig. 4. 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 FRAPTRAN input file. In what follows, the transient results of the fuel rods from these different zones (referred as T01-T06) will be discussed.

Table II lists the LBLOCA sequences of TRACE [1]. According to the setting in TRACE, the reactor scrammed at 0.5 sec while the reactor pressure reached 12.8 MPa. When the pressure has decreased below the safety injection (SI) setpoint (11.8MPa), low head safety injection (LHSI) and high head safety injection (HHSI) begin to be injected into the reactor coolant system with a delay time of 27sec.As the reactor coolant system pressure decreased below 4.24MPa, accumulator injection started. Figs. 5~7 show the power and boundary condition settings of FRAPTRAN which are offered by TRACE results during LBLOCA. “Heat” option was chosen as the boundary condition setting in FRAPTRAN.

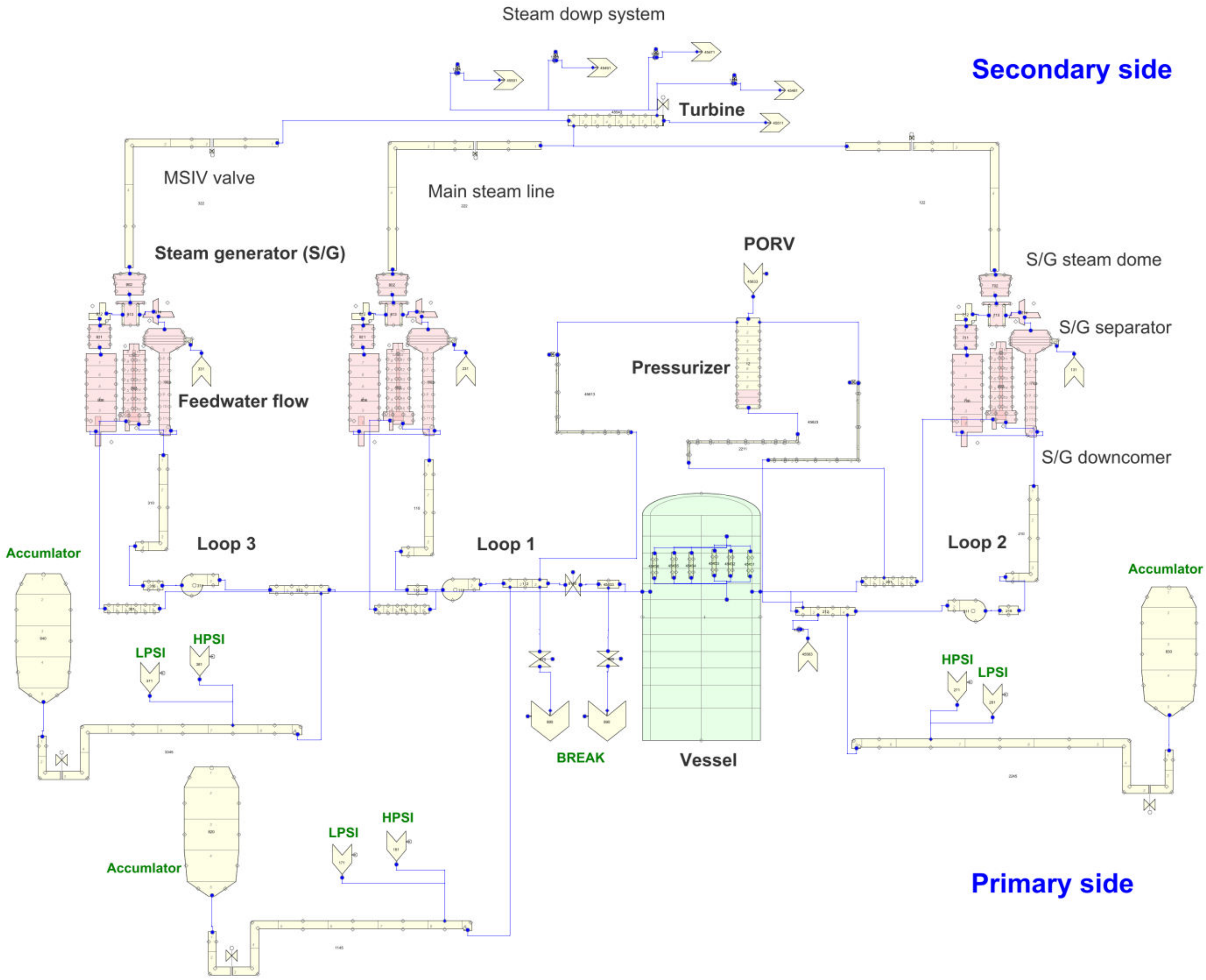

Fig. 1 Maanshan NPP TRACE model

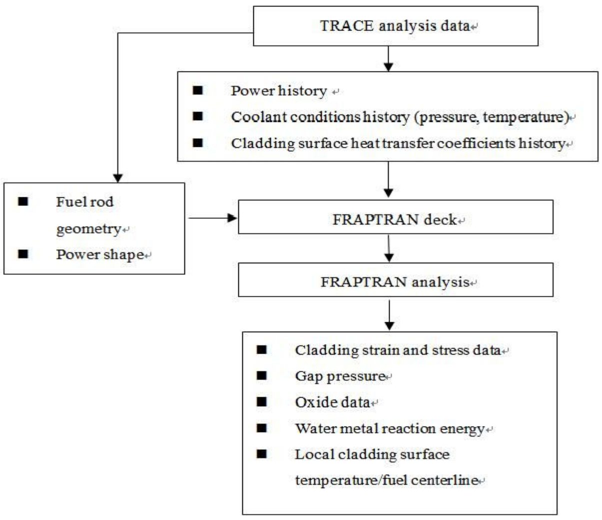

Fig. 2 The flowchart of combining TRACE and FRAPTRAN

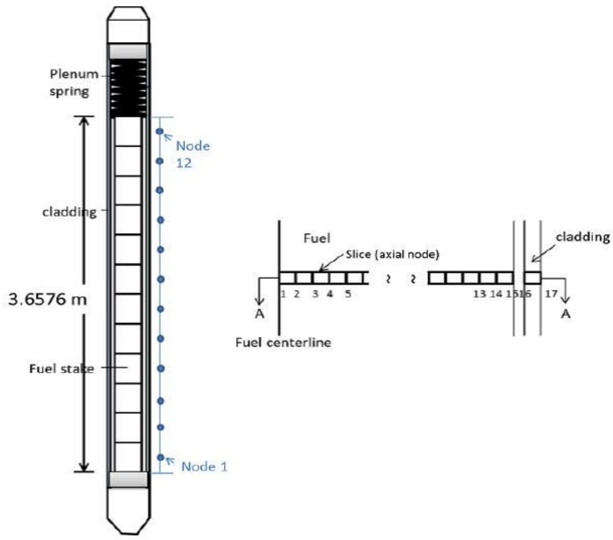

Fig. 3 The FRAPTRAN fuel rod geometry of Maanshan NPP

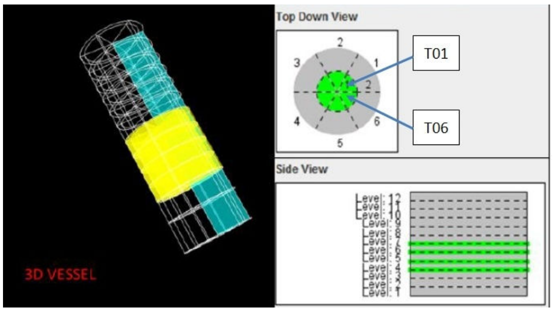

Fig. 4 The TRACE vessel structure model of Maanshan NPP

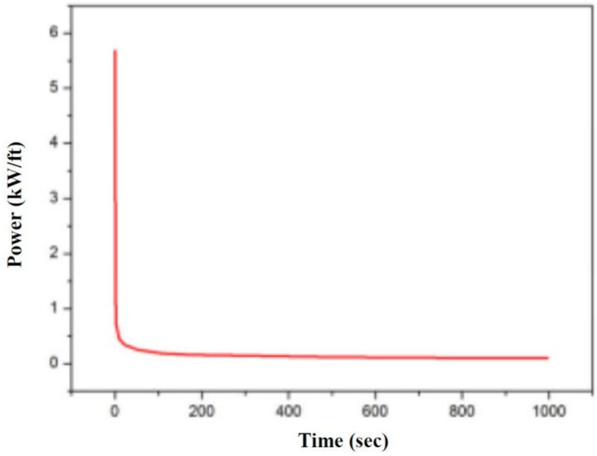

Fig. 5 The power data of TRACE

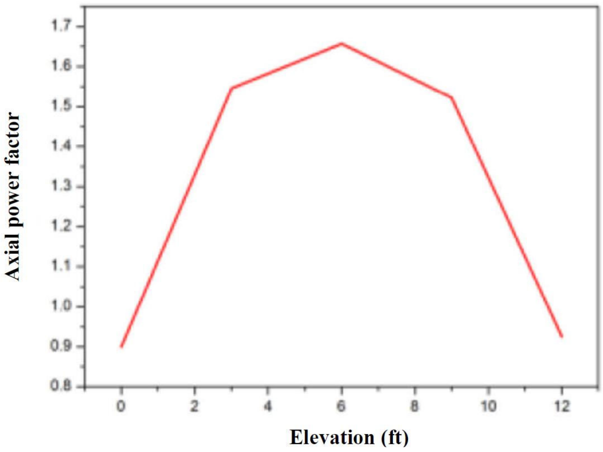

Fig. 6 The axial power shape data of TRACE

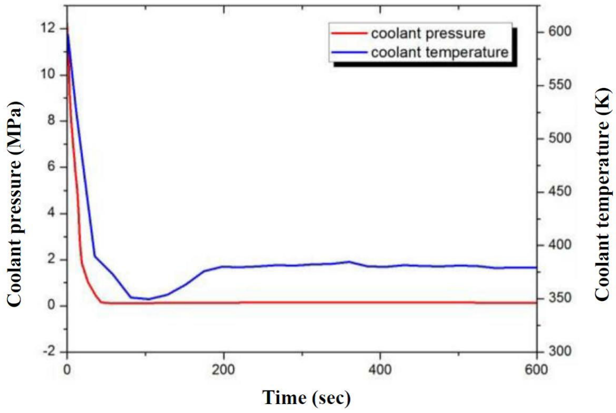

Fig. 7 The coolant pressure and temperature data of TRACE

TABLE I
FUEL ROD DESIGN PARAMETERS

Parameters	Values
Fuel active length(m)	3.6576
Fuel rod OD(m)	$9.14\text{x}10^{-3}$
Cladding type	ZIRLO
Cladding thickness(m)	$5.715\ \text{x}10^{-4}$
Filling gas	He(1.551MPa)

TABLE II
LBLOCA SEQUENCES OF TRACE DATA

LBLOCA events	TRACE data(sec)
Break began	0
Reactor scram setpoint reached (12.8 MPa)	0.5
SI signal generated (11.8 MPa)	1.5
Accumulators injection (4.24 MPa)	14.2

III. RESULTS

Fig. 8 shows the results of gap width between the fuel pellet and cladding. Because the core pressure drops and the coolant temperature decreases, the gap width increases after the LOCA occurs. The plenum pressure variation at T01 zone is shown in Fig. 9. Due to the gap width increase, the plenum pressure drops from 4.31 MPa to 2.2 MPa, and it appears that there is no failure of the fuel rod. Furthermore, the hoop stress simulation results of T03 zone are shown in Fig. 10 as an example. The hoop stress results of different nodes are nearly the same, and the maximum 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 Emergency Core Cooling System (ECCS) must be designed and activated so that peak cladding temperature (PCT) should not exceed 1477.6 K.

Fig. 11 shows the fuel centerline temperatures of nodes 1, 4, 7, 10, 12 in the T01 zone. The fuel centerline temperature is around 10 K higher than the outside cladding temperature. It's obvious that the peak cladding temperatures in six zones and the fuel centerline temperature were all below the criteria requirement. According to the present FRAPTRAN results for

International Journal of Engineering, Mathematical and Physical Sciences
ISSN: 2517-9934
Vol:8, No:1, 2014

the fuel rods at the six zones of 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 [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 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.

Finally, SNAP program provides the function of the animation presentation for TRACE/FRAPTRAN results. In the animation model, the parameters response of the Maanshan NPP and the performance of the fuel rod during LBLOCA can be presented (see Fig. 12).

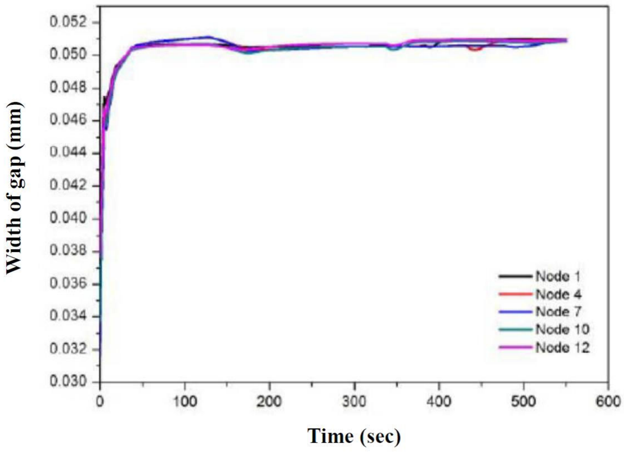

Fig. 8 The variations of gap width in the rod during transient at T03 zone

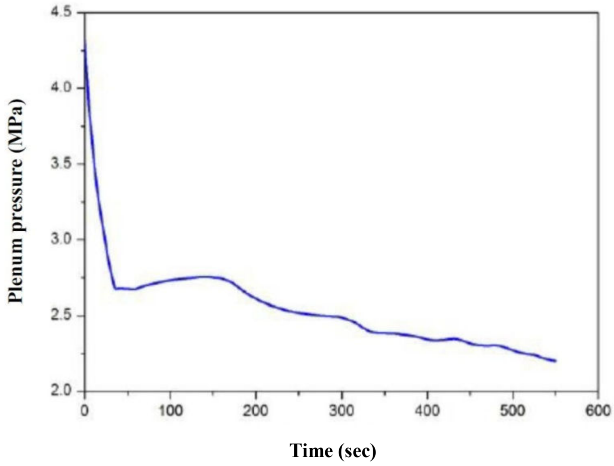

Fig. 9 The plenum pressure variation during transient at T01 zone

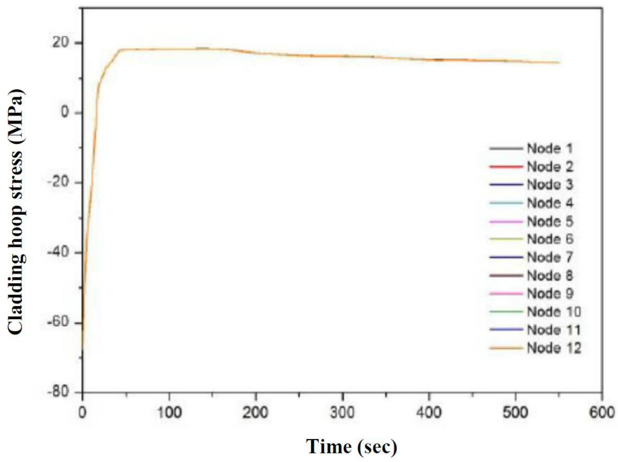

Fig. 10 The hoop stress variations during transient at T03 zone

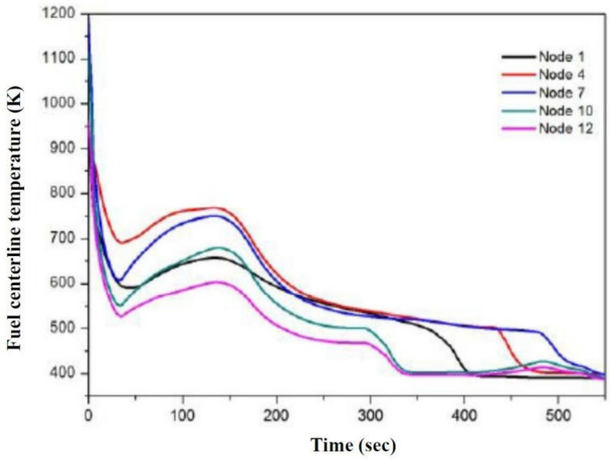

Fig. 11 The fuel centerline temperatures at T01 zone

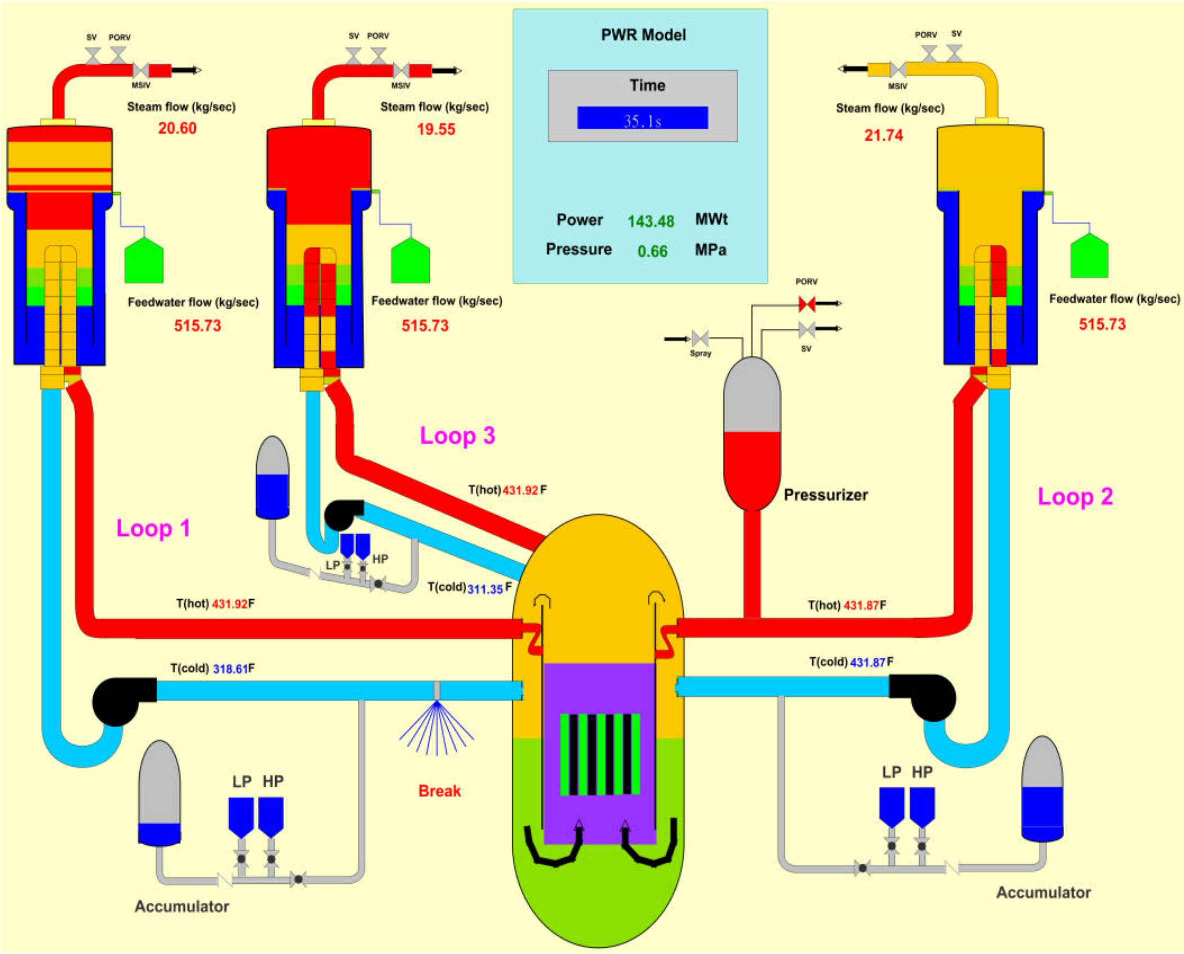

Fig. 12 The animation model of Maanshan NPP

IV. Conclusion

In this study, TRACE and FRAPTRAN under SNAP program were used to analyze the performance of the fuel rod for the LBLOCA transient of Maanshan NPP. Important results can be summarized as follows:

1. TRACE and FRAPTRAN have been successfully combined to analyze the fuel rod transient behaviors in the LBLOCA scenario.
2. Based on the calculation results of plenum pressure and the gap thickness, there was no cladding ballooning or failure in the present case.
3. The maximum hoop stress was about 18 MPa and the oxide thickness was 0.003mm for the present simulation cases, which are all within the safety operation range.
4. Based on the analysis results, the cladding temperatures and the fuel centerline temperatures were all below the criteria requirement, which ensure a safety operation condition for the present case.

References

[1] J. H. Yang, J. R.Wang and H. T.Lin, C. Shih, “LBLOCA analysis for the Maanshan PWR nuclear power plant using TRACE”, 2011 2nd International Conference on Advances in Energy Engineering (ICAEE) Vol. 14, pp. 292–297, 2012.
[2] U.S. NRC, “TRACE V5.0 user manual,”2010.
[3] U.S. NRC, “U. S. Code of Federal Regulations, Title 10, Energy, Parts0 to 50,”1997.
[4] K. Geelhood, “Modeling high burnup RIA tests with FRAPTRAN,” Proceedings of 2010 LWR Fuel Performance, 2010.
[5] A. Daavittila, A. Hämäläinen and H. Räty, “Transient and fuel performance analysis with VTT’s coupled code system,” Mathematics and Computation, Supercomputing, Reactor Physics and Nuclear and Biological Applications, 2005.
[6] K. J. Geelhood, W. G. Luscher, C. E. Beyer and J. M. Cuta, “FRAPTRAN 1.4: a computer code for the transient analysis of oxide fuel rods,” NUREG/CR-7023, Vol. 1, 2011.
[7] K.J. Geelhood, W.G. Luscher, C.E. Beyer and J.M. Cuta, “FRAPTRAN 1.4: integral assessment,” NUREG/CR-7023, Vol. 2, 2011.
[8] Y. Lee, T.J. Mckrell and M. S. Kazimi, “Thermal shock fracture of silicon carbide and its application to LWR fuel cladding performance during reflood,” Int. Congress on Advances in Nuclear Power Plants (ICAPP ’13), Jeju Island, Korea, April 14-18, 2013 Paper No. FA170, 2013.
[9] Y. S. Bang, A. J. Cheong and S. W. Woo, “Multi-dimensional thermal-hydraulic response and uncertainty of calculation for LBLOCA of APR1400,”Int.Congress on Advances in Nuclear Power Plants (ICAPP ’13),Jeju Island, Korea, April 14-18, 2013 Paper No. FA002, 2013.