International Journal of Chemical, Materials and Biomolecular Sciences
ISSN: 2415-6620
Vol:8, No:7, 2014

Turbine Trip without Bypass Analysis of Kuosheng Nuclear Power Plant Using TRACE Coupling with FRAPTRAN

J. R. Wang, H. T. Lin, H. C. Chang, W. K. Lin, W. Y. Li, C. Shih

Abstract**—**This analysis of Kuosheng nuclear power plant (NPP) was performed mainly by TRACE, assisted with FRAPTRAN and FRAPCON. SNAP v2.2.1 and TRACE v5.0p3 are used to develop the Kuosheng NPP SPU TRACE model which can simulate the turbine trip without bypass transient. From the analysis of TRACE, the important parameters such as dome pressure, coolant temperature and pressure can be determined. Through these parameters, comparing with the criteria which were formulated by United States Nuclear Regulatory Commission (U.S. NRC), we can determine whether the Kuoshengnuclear power plant failed or not in the accident analysis. However, from the data of TRACE, the fuel rods status cannot be determined. With the information from TRACE and burn-up analysis obtained from FRAPCON, FRAPTRAN analyzes more details about the fuel rods in this transient. Besides, through the SNAP interface, the data results can be presented as an animation. From the animation, the TRACE and FRAPTRAN data can be merged together that may be realized by the readers more easily. In this research, TRACE showed that the maximum dome pressure of the reactor reaches to 8.32 MPa, which is lower than the acceptance limit 9.58 MPa. Furthermore, FRAPTRAN revels that the maximum strain is about 0.00165, which is below the criteria 0.01. In addition, cladding enthalpy is 52.44 cal/g which is lower than 170 cal/g specified by the USNRC NUREG-0800 Standard Review Plan.

Keywords**—**Turbine trip without bypass, Kuosheng NPP, TRACE, FRAPTRAN, SNAP animation.

I. Introduction

THE safety analysis of the nuclear power plant (NPP) is very important work especially after the Fukushima NPP event occurred. From the accident at the Japanese Fukusima NPP, an extreme event beyond the design basis is realized to be possible. As a result, the more severe hypothetical accident situation should be concerned. In this trend, the Kuosheng NPP (located on the northern coast of Taiwan) has been done a series of severe hypothetical accident analysis and this study, the turbine trip without bypass, is one of these hypothetical accident analyses. The nuclear steam supply system (NSSS) of Kuosheng NPP is a type of BWR/6 reactor, 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. Unit 1 has started Stretch Power Uprate (SPU) from Cycle 24 and Unit 2 has started SPU from Cycle 23. The operating power is 104.7% of the OLTP (Original Licensed Thermal Power), reaching to the value 3030 MWt now [1].

TRACE, the TRAC/RELAP Advanced Computational Engine developed by U.S. NRC, has been designed to perform best-estimate analysis of loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in reactor systems. Models used include multidimensional two-phase flow, nonequilibrium thermo-dynamics, generalized heat transfer, reflood, level tracking, and reactor kinetics. Automatic steady-state and dump/restart capabilities are also provided [2].

Fuel Rod Analysis Program Transient code (FRAPTRAN) is conducted by the U.S. NRC and developed by Pacific Northwest National Laboratory (PNNL). It is an analytical tool that calculates LWR fuel rod behavior when power or coolant boundary conditions, or both, are rapidly changing. In this study, the FRAPTRAN code provides more fuel rod details which are discussed with the TRACE results to obtain a more accurate analysis report [3].

In these recent years, a powerful user interface program named Symbolic Nuclear Analysis Package (SNAP) is widely applied to simplify the process of performing engineering analysis. It supports several nuclear analysis codes such as TRACE, PARCS, FRAPCON, FRAPTRAN etc. In this study, both the Kuosheng NPP TRACE and FRAPTRAN model were built through the SNAP interface. It simplifies the model building and data input process. Moreover, getting the data from the SNAP interface model, users can further make animations from these results to illustrate the variation of transient states more clearly [4].

II. Methodology and Modeling

A. The TRACE Code

TRACE is a modernized code with the capability to simulate the reactor system and model the thermal-hydraulic phenomena in three-dimensional space. This program is a component-based code for fast and integrated inputs of reactor systems. The reactor vessel, fuel bundles, separators, dryers and jet pumps are modeled by the specific components such as VESSEL, CHAN, SEPD and JETPUMP respectively. Moreover, in this study, TRACE is integrated into SNAP to develop TRACE input decks and NPP model more quickly and conveniently. Instead of those out-of-date codes like TRAC and

J. R. Wang and H.T. Lin are with the Institute of Nuclear Energy Research, Atomic Energy Council, R.O.C., 1000, Wenhua Rd., Chiaan Village, Lungtan, Taoyuan, 325, Taiwan.

H.C. Chang is with the Institute of Nuclear Engineering and Science, National Tsing-Hua University, 101 Section 2, Kuang Fu Rd., HsinChu, Taiwan; (Phone: +886-913-325171; e-mail: ms9831117@gmail.com).

W.K. Lin, W.Y. Li and C. Shih are with the Institute of Nuclear Engineering and Science, National Tsing-Hua University, 101 Section 2, Kuang Fu Rd., HsinChu, Taiwan.

RELAP, the combination of TRACE and SNAP provides a new man-machine interface and becomes a powerful thermal-hydraulic analysis tool. In this research, SNAP v2.2.1 and TRACE v5.0p3 are employed.

B. TRACE Model of Kuosheng NPP

The Kuosheng NPP TRACE model (Fig. 1) has been built according to the FSAR, design documents, and TRACE manuals [1], [2], [5]. In this model, the reactor, which is divided into two azimuthal sectors, four radial rings, eleven axial levels, altogether eighty-eight cells, is simulated by the 3-Dvessel component (Fig. 2). 10 groups of injection pumps are merged into an equal injection pump. Two recirculation loops are set outside the reactor, with a recirculation pump in each loop. Several valve components are built to simulate the main steam line isolation valves (MSIVs), turbine stop valves (TSVs), turbine control valves (TCVs) and safety relief valves (SRVs). The steam goes through the top of the reactor and then enters the main steam lines. Finally, the steam passes through the TCVs and drives the turbines (boundary conditions). We also build bypass pipelines and the turbine bypass valve. The break component at the end of bypass valve is used to simulate the condenser. In the Kuosheng NPP TRACE model, there are three simulation control systems included (1) feed water flow control system, (2) steam bypass and pressure control system and (3) recirculation flow control system. Currently, these three control systems have been built by the signal variables, control blocks, trips and other components of SNAP/TRACE. Besides, in Kuosheng NPP TRACE model, "point kinetic" parameters such as delay neutron fraction, Doppler reactivity coefficient, and void reactivity coefficient are provided as TRACE input for power calculations. For the six channels, they are one dimensional components divided into 10 cells with 10 rods per row. In each row, it contains varying from 96 to 120 fuel bundles. Fig. 3 shows the arrangement of fuel rods in each channel.

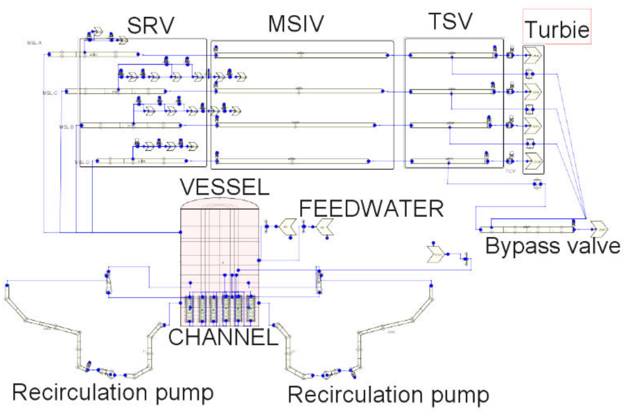


Fig. 1 TRACE model of Kuosheng nuclear power plant

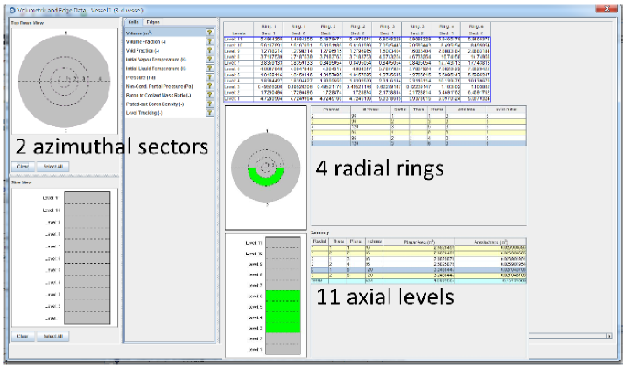


Fig. 2 Vessel geometry of the TRACE model

C. Set Points of the Hypothetical Accident in TRACE Model

In this study, the hypothetical turbine trip accident is divided into two parts. The first 500 seconds period is a steady state analysis with initial conditions of power 3030MWt and core flow 11177.3 kg/sec. After the 500 seconds period, the turbines failed and gave a signal to cause the closure of turbine stop valves (TSVs). In general, once the TSVs start to close, the turbine bypass valves start to open to relieve the pressure of the main steam line. However, in this case, to simulate a more severe situation, the bypass valves do not open. The main steam line pressure increases continually until the safety relief valves (SRVs) open and relieve the high pressure steam.

The reactor scram and recirculation pump trip (RPT) are initiated when the TSVs reach 90% open. Furthermore, only six safety relief valves (SRVs), which will be turned on at 7.94MPa and turned off at 7.63MPa, are available in this case. Table I and II list more details of initial conditions and set points.

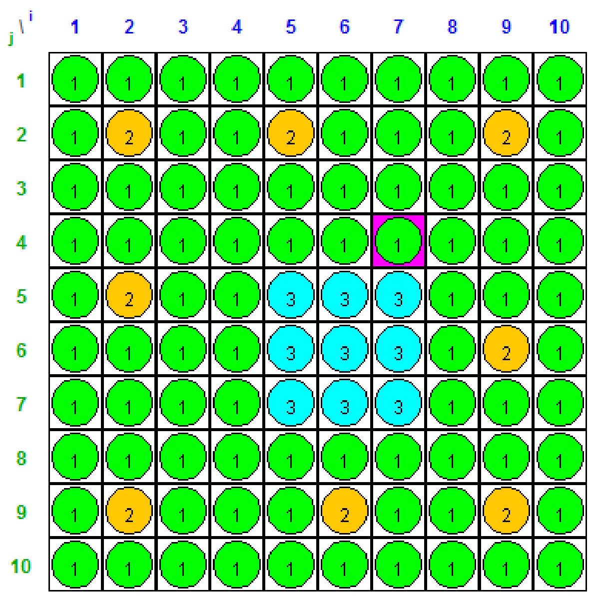


Fig. 3 Rod arrangement of the six channels in the vessel

TABLE I
SET POINT OF TURBINE TRIP WITHOUT BYPASS MODEL

Action	Set point
TSV closure	0 sec
Reactor scram	90% TSV open
TSV fully closure	0.1 sec
Control rods insert	0.16 sec
SRVs open	Dome pressure reach to 7.94MPa

TABLE II
INITIAL CONDITION AND STEADY STATE RESULTS OF TRACE

Parameters	TRACE data
Power (MWt)	3030
Dome Pressure (MPa)	7.17
Feedwater Flow (kg/sec)	1641
Steam Flow (kg/sec)	1641
Core inlet flow (kg/sec)	10705
Feedwater Temperature (K)	488.7

D. The FRAPTRAN Code and Modeling

In the TRACE model, the fuel bundles are built by channels combined with power components which can simulate the fuel rod heat generation. TRACE can only provide the temperature distribution during the transient state to determine whether the fuel rods failed or not. Hence, FRAPRAN is a good tool to supply details of fuel rods and confirm the results of TRACE.

To carry on this hypothetical analysis, in addition to the geometry and the fuel rod power history of the fuel rod, FRAPTRAN still needs heat transfer coefficient, coolant pressure and temperature information from the TRACE result. In this study, the power history from the TRACE model is the summation of all fuel rods. It should be evenly distributed over fuel rods before entering the FRAPTRAN model.

The fuel rod was divided into 12 nodes in equal space. Node 1 to 12 stands for the different positions from the bottom to the top respectively (Fig. 4). Fig. 5 shows the appearance of FRAPTRAN model in the SNAP interface. In the left side of Fig. 5 are parameters that should be entered according to the TRACE data results; on the other hand, the right side of this figure is the job stream of the FRAPTRAN model.

To simulate the fuel rod more accurately, the burn-up information obtained from FRAPCON was also concerned. In this study, it was assumed that the fuel rods has burned for 18 months and the burn-up value of FRAPCON result is 17GWd/MTU.

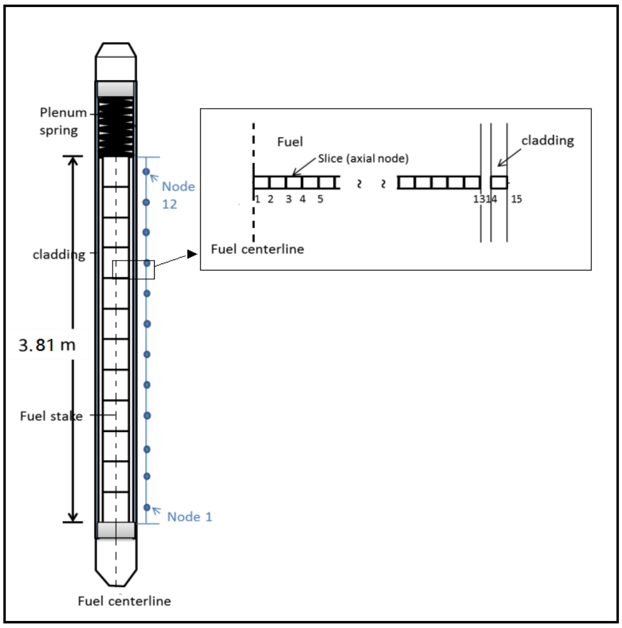


Fig. 4 Fuel rod geometry of the FRAPTRAN model

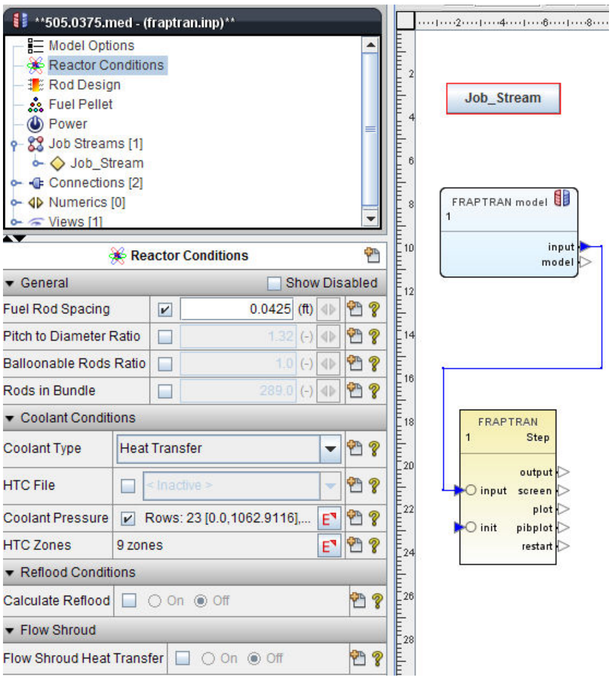


Fig. 5 FRAPTRAN model of the SNAP interface

E. SNAP Animation Modeling

Fig. 6 shows the animation appearance of the Kuosheng NPP in this hypothetical accident. In this animation model, the reactor vessel was built by two plenum components. The dome pressure data was showed by the plenum components, which were connected with the top vessel level. In the middle of the reactor vessel were three axial components which show the data results of FRAPTRAN model. The left axial component represents the temperature of the fuel rod; the middle one

International Journal of Chemical, Materials and Biomolecular Sciences
ISSN: 2415-6620
Vol:8, No:7, 2014

shows the enthalpy results and the right one was the strain variation of the fuel rod in this hypothetical accident. There are 4 main steam lines in the reactor of Kuosheng NPP. However, to simplify the animation and make the appearance clearer, there is only one main steam line and TSV showed in this animation at the right side of the vessel. In the same reason, there is only one SRV and pipe which represented 6 available SRVs of this case. At the left side of the vessel, there feedwater filling system and at the lower left corner of the animation model is a recirculation pump. In addition, there are several blocks that show the time sequences. When the specific step has reached, the background color transforms from white to yellow.

Besides the combination animation model of the TRACE and FRAPTRAN results mentioned above, another animation model which only contains the FRAPTRAN results is built to illustrate the variation of the fuel rod more clearly. Fig. 7 is the appearance of the FRAPTRAN analysis animation model. In the model, there are 3 fuel rods which stand for acceptance limits including temperature, enthalpy and strain respectively. There are 3 color maps next to fuel rods respectively to indicate the variation of different parameters. In addition, the 3-D graph which array were set to be (1,1) are built beside the fuel rods to reveal the greatest value of the fuel rod, which is node 8 in this case. At the upper right corner is the time sequence of this transient state analysis.

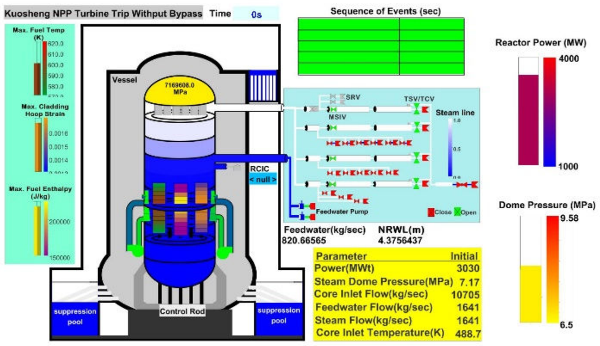


Fig. 5 Combination animation appearance of TRACE and FRAPTRA

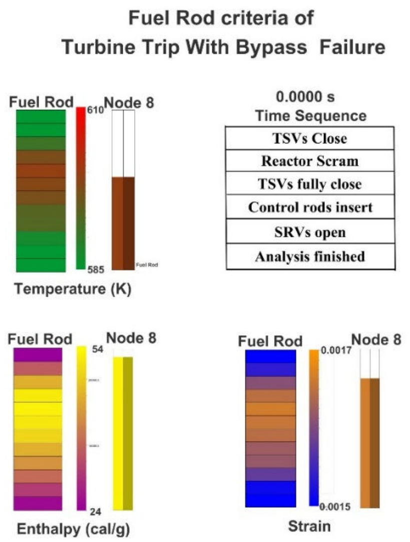


Fig. 6 Animation appearance of FRAPTRAN model only

III. Result and Discussion

A. TRACE Results and Discussion

To plot the data results more clearly, the data of the first500 seconds steady state are removed. All of the parameters in the figure are plotted from the beginning of the TSVs closure. In these figures below, they only illustrate the last 5 seconds transient state of the TRACE model.

At the beginning of TSV closure, main steam line vapor flow start to decrease immediately. Hence, the vapor is collected on the top of the vessel, which caused the increase of the dome pressure (Fig. 8). At the same time, because of the dome pressure increase, the void fraction may be affected and start to decline. This declination made the neutron in the reactor core get a positive reactivity and hence increase the core power (Fig. 9). After that, the core power began to decrease because of the reactor scram. Though the reactor had been shut down when the TSV reach to 90% open, it needed some time to decline the core power; as a result, core power declined at time point 1 second (Fig. 10).

On the other hand, as the core power start to raise up, the main steam line flow increased again until the core power reach to the peak value at about 1.2 second; then, it declined again with the core power (Fig. 10).

At 1.2 second, the dome pressure reach to 7.94 MPa, the set point of SRVs initiated. However, from the tendency in Fig. 11, the SRVs did not open immediately because the six available SRVs are set with a lag time 0.4 second [6]. As a result, the dome pressure continually increased until the value of about 8.2 MPa, which did not exceed the criteria 9.58 MPa yet. After the 0.4 second lag time, the SRVs open and the dome pressure decreased (Fig. 11). The main steam line flow got a relatively steady value (Fig. 12). This value is about half of the initial flow rate because the recirculation pump was tripped and the feedwater control system reduced the injection of coolant water. From these data above, we can learn every details and steps of the turbine trip without bypass transient case. To the end of the transient analysis, the dome pressure did not increase again and the peak value was not over the acceptance limit 9.58MPa. It indicates that the Kuosheng NPP is safe in this TRACE analysis model.

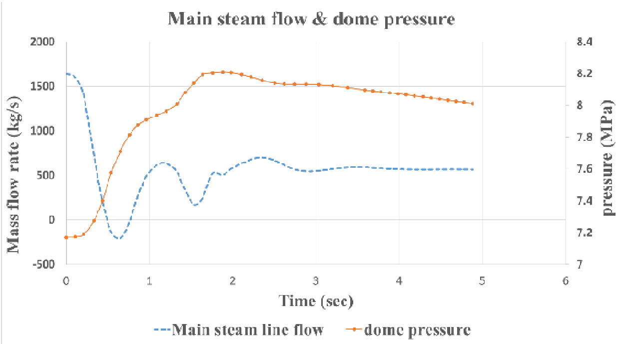


Fig. 7 TSVs closed, main steam line flow decrease and dome pressure raised immediately

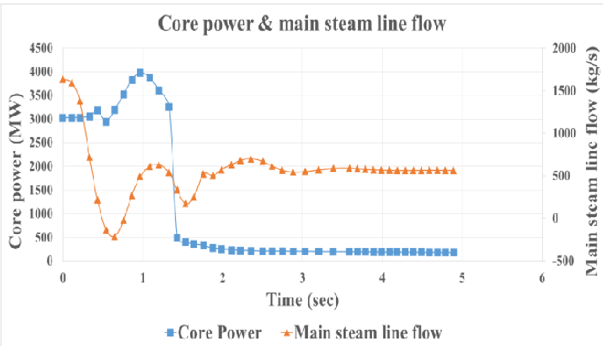


Fig. 8 Core power raise due to the void fraction declination

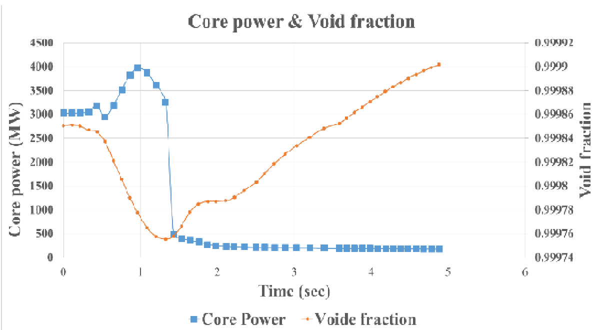


Fig. 9 Main steam line flow increase again due to the increase of core power

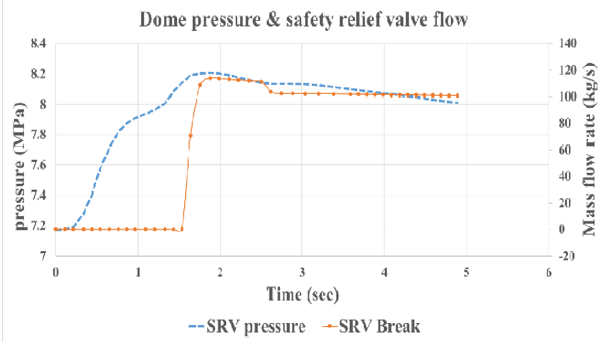


Fig. 10 SRVs obtained open as the dome pressure reached to 7.94MPa with 0.4 second delay

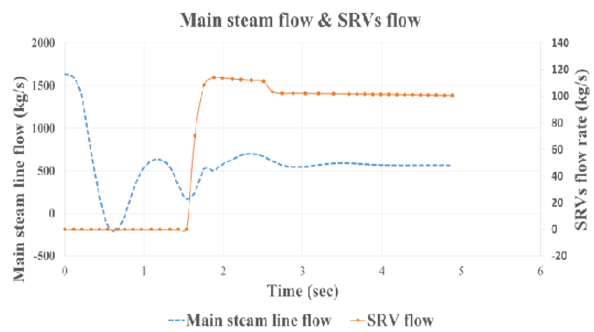
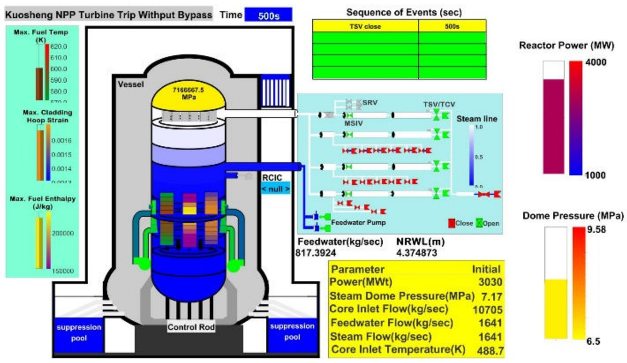
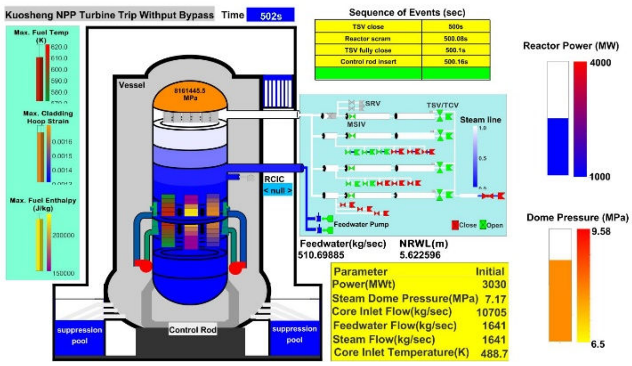


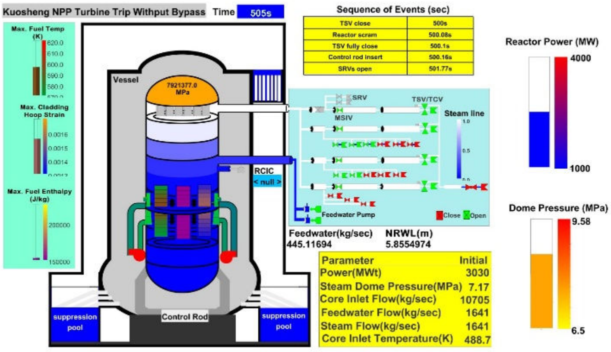
Fig. 11 SRVs open and the main steam line flow reached to a steady state



(a)



(b)



(c)

Fig. 12 (a) TSVs closed and dome pressure start to increase. (b) Dome pressure reached to 7.94MPa and SRVs open. (c) Analysis finished and all the parameters of Kuosheng NPP did not exceed the acceptance limits

Applying the animation model to the TRACE data results, the strip plots above can be assembled together and be exported as a video, which could be understood more easily for readers. Fig. 6 is the steady state diagram of the hypothetical accident at about 499 second. In this figure, steam produced by the reactor core goes through the main steam line and both the feed water pump and recirculation pump worked normally. As the TSVs are closed (Fig. 13 (a)), the steam released nowhere and the reactor vessel pressure increased immediately. Once the dome pressure reached to the value 7.93 MPa, the SRVs would obtain the open signal. After 0.4 second lag time, the SRVs open and the steam can go through the relief valves to decrease the pressure inside the vessel (Fig. 13 (b)). In the end of the analysis, all the parameters including dome pressure, main steam line flow, feedwater and recirculation flow reached to steady values (Fig. 13 (c)).

B. FRAPTRAN Results and Discussion

For the convenience, node 8, which has the extremely values, is chosen to be discussed and plotted in figures below. The sudden change of power due to the variation of void fraction was the analysis result from the TRACE model. It directly (but not immediately) affected the cladding temperature (Fig. 14). The lag time came from the conduction of heat from the fuel pellets to the cladding.

In the FRAPTRAN deformation model, the hoop strain is divided into two segments, including thermal hoop strain and elastic hoop strain. Cladding temperature dominates the thermal hoop strain. Fig. 15 illustrates the relationship between cladding temperature and thermal hoop strain. The trend of thermal strain is very similar to that of cladding temperature, which means that the thermal strain is very sensitive with the temperature.

On the other hand, the elastic hoop strain is related with the hoop stress, which is the additive effect of the gap gas pressure and the coolant pressure. In this case, the coolant pressure (outside the cladding) is greater than the gap gas pressure (inside the cladding), which means that the cladding felt a compressive stress. As a result, the elastic hoop strain was negative. Though the pressure outside the fuel rod is greater than that inside the fuel rod, the cladding still expanded before the time 1.2 second. Because the thermal strain value is much greater than elastic strain value, the total hoop strain is still positive and the cladding expanded. After 1.2 second, the cladding temperature started to decrease because of the declination of the cladding temperature. Furthermore, the hoop stress had reached to a steady value from the time 2 seconds; that is, from this moment, the cladding temperature (thermal hoop strain) dominated the cladding deformation. The cladding decreased and the cladding started to shrink. The hoop strain peak value 0.00166 occurred at time 1.2 second, much less than the criteria 0.01. Furthermore, the cladding enthalpy has the peak value about 52 cal/g (shown in Fig. 16), which is less than the criteria 170 cal/g. Both values indicated that the fuel rod kept safe in the turbine trip without bypass hypothetical accident.

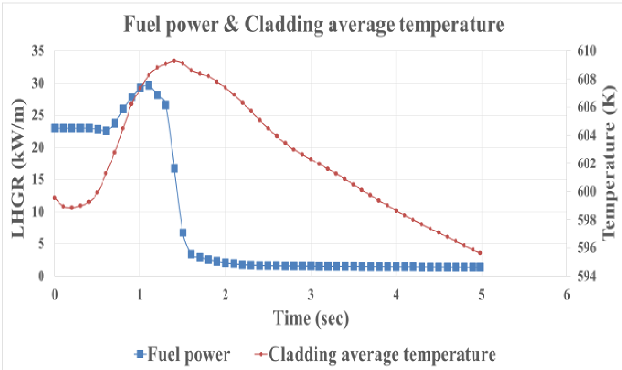


Fig. 13 The core power directly affected the cladding temperature

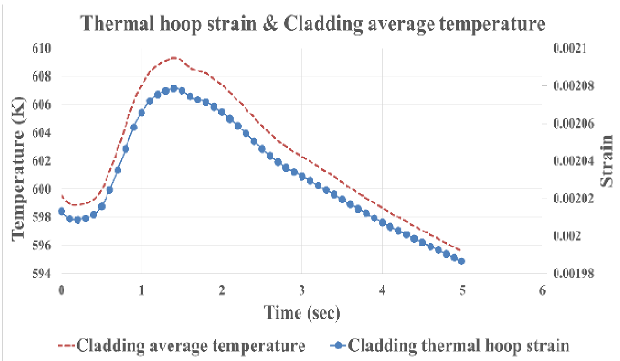


Fig. 14 Thermal hoop strain has a strong relationship with cladding temperature

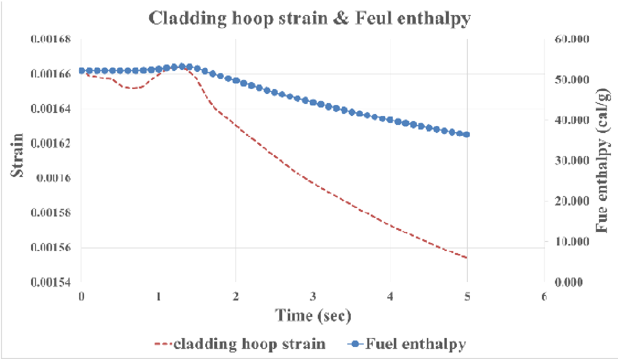
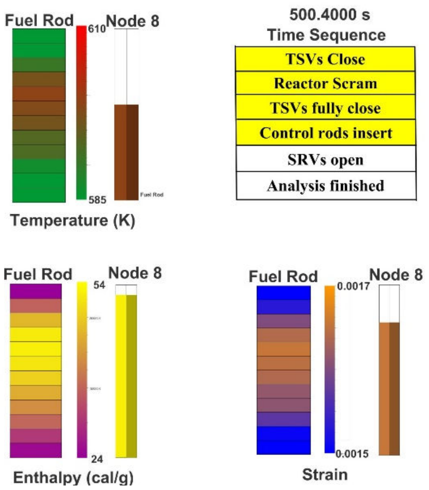
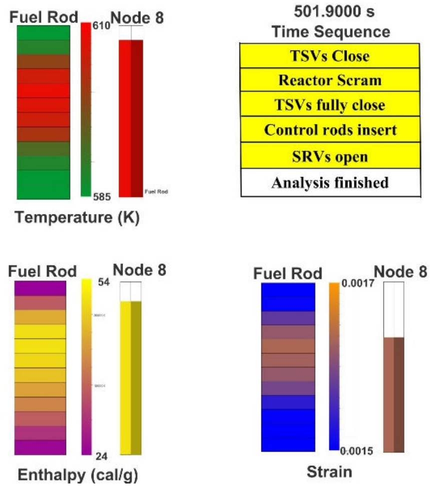


Fig. 15 Both the cladding hoop strain and enthalpy did not exceed the acceptance limit

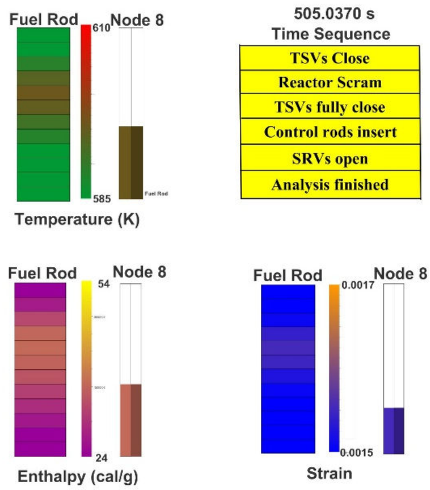
In addition to the TRACE-FRAPTRAN combination model, the data results of FRAPTRAN were also depicted in the FRAPTRAN animation model. At the time point of TSVs closure and reactor scram, the fuel rods have no obvious change. After the time point of 500.4 second, the enthalpy and strain start to increase together with the temperature (Fig. 16 (a)). As the SRVs open at about 501.9 second, all these 3 parameters start to decrease till the analysis finished (Figs. 16 (b) and (c)).



(a)



(b)



(c)

Fig. 16 (a) All these 3 parameters start to raise at the time point 500.4 second. (b)SRVs open and these 3 parameters start to decrease. (c)At end of the analysis, these three parameters reach back to steady values and the fuel rod still kept its integrity

IV. Conclusions

According to the comparison results of TRACE and INER report data, it indicates that there is a respectable accuracy in the Kuosheng NPP SPU TRACE model. With this model, the turbine trip without bypass analysis results indicate that the maximum vessel pressure is below the acceptance limit of 9.58 MPa. Furthermore, this thermal hydraulic model was integrated with the fuel rods analysis, which can provide more details of a single rod. Comparing those parameters of a single fuel rod with the criteria, FRAPTRAN depicts that the integrity of fuel rods are still kept. Different from previous studies, the data results of turbine trip without bypass case can be illustrated through the animation model, which can present many data results of the transient state in the same time. Moreover, in the future studies which are similar with the turbine trip without bypass analysis, the users may apply mechanically of the animation model of this case.

References

[1] Taiwan Power Company, "Final Safety Analysis Report for Kuosheng Nuclear Power Station Units 1&2 (FSAR)", 2001.

[2] U.S. NRC, "TRACE V5.0 user's manual", Division of Safety Analysis Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission Washington, DC 20555-0001, 2010.

[3] K.J. Geelhood, W.G. Luscher, C.E. Beyer, "FRAPTRAN 1.4: Integral Assessment", Pacific Northwest National LaboratoryP.O. Box 999 Richland, WA 99352.

[4] U.S. NRC, "Symbolic Nuclear Analysis Package (SNAP) User's Manual", Applied Programming Technology, Inc. 2007

[5] INER, "Guideline of Generating Parameters for Reload Licensing Analyses for Kuosheng Units 1 and 2", INER-6529R, 2009.

[6] Taiwan Power Company, "Technical Specifications Revision 0, Kuosheng Nuclear Power Station Units 1 & 2", January 2008.