The establishment and application of TRACE/FRAPTRAN/SNAP methodology for Kuosheng nuclear power plant

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Abstract: Kuosheng Nuclear Power Plant (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. First, Kuosheng NPP TRACE/SNAP model was developed in this research. In order to assess the system response of the Kuosheng NPP TRACE/SNAP model, this study used startup tests data to evaluate the Kuosheng NPP TRACE/SNAP model. Second, the transient analysis of Kuosheng NPP TRACE/SNAP model under the more severe conditions was performed. Besides, in order to confirm the mechanical property and integrity of fuel rods, the FRAPTRAN analysis was also performed in this study.

Keyword: TRACE; safety analysis; BWR/6; severe accident

1 Introduction

The advanced thermal hydraulic code named TRACE has been developed by U.S. NRC for NPP safety analysis. According to TRACE manual [1], one of the features of TRACE is its capacity to model the reactor vessel with 3-D geometry. It could support a more accurate and detailed safety analysis for nuclear power plants. TRACE has the greater simulation capability than other old codes (TRAC-P, TRAC-B, RELAP5 and RAMONA), especially for events such as LOCA. Besides, a graphic user interface program, SNAP, which processes inputs, outputs, and the animation model for TRACE, has also been developing.

FRAPTRAN is a Fortran language computer code that calculates the transient performance of light-water reactor fuel rods during reactor transients and hypothetical accidents such as loss-of-coolant accidents, anticipated transients without scram, and reactivity-initiated accidents [2]. According to FRAPTRAN manual [3], FRAPTRAN code was verified by the experimental data of IFA-650, MT-1, MT-4, LOC-11C, and TREAT. The above comparison results indicate that FRAPTRAN is capable of handling fuel rods analysis.

Kuosheng NPP's 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.

In our previous research [4]-[9], we established Maanshan NPP (PWR), Chinshan NPP (BWR/4), and Lungmen NPP (ABWR) TRACE/SNAP models successfully. Subsequently, based on the successful experience from the above models, this research focuses on the establishment of the Kuosheng NPP TRACE/SNAP model. Kuosheng NPP TRACE/SNAP model included one 3-D vessel, six channels which are used to simulate 624 fuel bundles, four steamlines, and 16 SRVs components, *etc.* In Maanshan NPP, Chinshan NPP, and Lungmen NPP TRACE/SNAP models, the containment and suppression pool were

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not simulated. But in Kuosheng NPP TRACE/SNAP model, we simulated the containment and suppression pool. In order to check the system response of Kuosheng NPP TRACE/SNAP model, we used startup tests data to assess the Kuosheng NPP TRACE/SNAP model. The load rejection and a feedwater pump trip transients were selected to validate Kuosheng NPP TRACE/SNAP model.

Besides, in order to estimate the safety of Kuosheng NPP under the more severe conditions, the SBO + LOCA transient analysis of Kuosheng NPP TRACE/SNAP model was performed which included the no water injection case and fire water injection case. Finally, TRACE's analysis results (ex: power and coolant conditions) were used in FRAPTRAN's input files. FRAPTRAN can calculate the cladding temperature, hoop stress/strain, oxide thickness of cladding of the fuel rods. Besides, the animation model of Kuosheng NPP was presented using the animation function of SNAP with TRACE analysis results.

2 Methodology

SNAP v 2.2.1 and TRACE v 5.0p3 were used in this research. Kuosheng NPP TRACE/SNAP model (Fig. 1) has been built according to the FSAR, design documents, and TRACE manuals [1],[10]-[13]. Kuosheng NPP reactor was simulated by the 3-Dvessel component which was divided into two azimuthal sectors, four radial rings, and eleven axial levels. Six channels (one dimensional component) were used for simulating 624 fuel bundles. Full length fuel rods, partial length fuel rods and water rods were also simulated in the channel components. Two recirculation loops were set outside the reactor, with a recirculation pump in each loop. 10 groups of jet pumps were merged into an equal jet pump. Four steam lines connected with the vessel and each steam line had one MSIV (main steam line isolation valve), several SRVs (safety relief valves), one TCV (turbine control valve), and one TSV (turbine stop valve). The bypass valve (BPV) was also simulated in this mode. We used valve components to simulate MSIV, SRVs, TCV, TSV and BPV. The critical flow models for MSIVs, SRVs, TCVs, TSVs, and BPV had been considered in our analysis. The containment of Kuosheng NPP was also simulated in the TRACE/SNAP model. The containment was composed of drywell, wetwell, suppression pool, vent annulus, horizontal vent, upper pool, and reactor building which were shown in Fig. 1. In Kuosheng NPP TRACE/SNAP model, there were three simulation control systems: (1) feed water flow control system, (2) steam bypass and pressure control system and (3) recirculation flow control system. Besides, in Kuosheng NPP TRACE/SNAP model, "point kinetic" parameters such as delay neutron fraction, Doppler reactivity coefficient, and void reactivity coefficient were provided as TRACE input for power calculations. Finally, SNAP used the TRACE results data to make an animation for the transient analysis, such as Fig. 2.

The geometry data of the fuel rod and the results from TRACE analysis (fuel rod power, coolant pressure, heat transfer coefficient) were inputted into FRAPTRAN to analyze the reliability of fuel rod. In FRAPTRAN model (see Fig. 3), node 1 is the bottom of the fuel rod and node 23 is the top of the fuel rod.

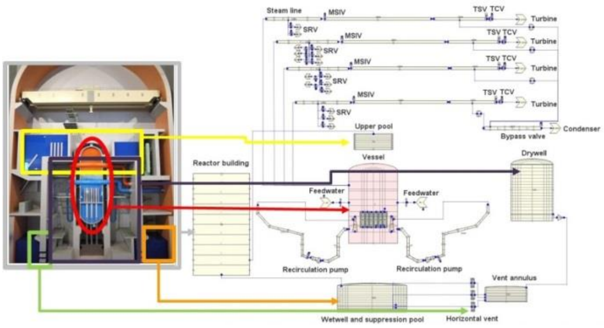


Fig. 1 TRACE/SNAP model of Kuosheng NPP.

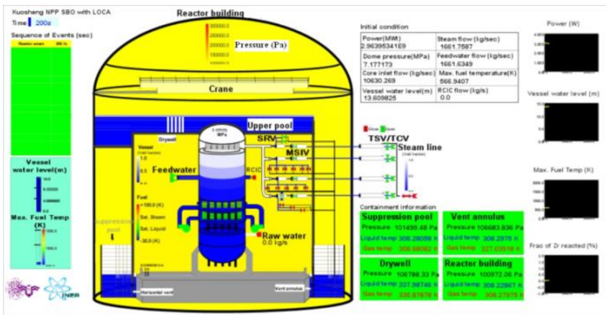


Fig. 2 SNAP animation model of Kuosheng NPP.

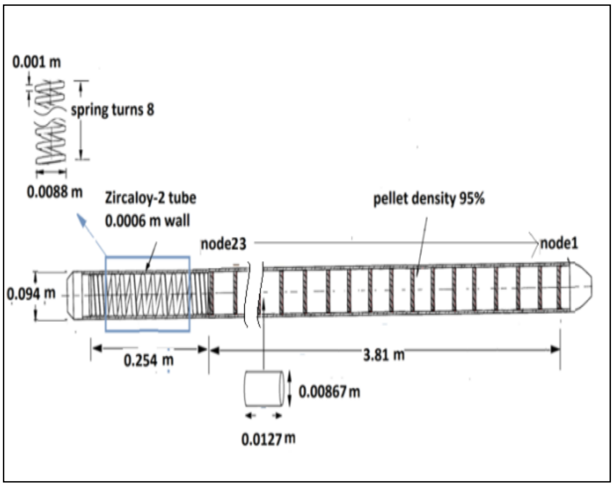


Fig. 3 FRAPTRAN model of Kuosheng NPP.

3 Results

Before the transient calculation of Kuosheng TRACE/SNAP model begins, it is necessary to carry out the steady state calculation and make sure that the system parameters (such as the feedwater flow, steam flow, dome pressure, and core flow, *etc.*) are in agreement with startup tests data under the steady state condition. The results of analysis of TRACE were clearly consistent with startup tests data under steady state conditions (See Table 1).

Table 1 The comparison of initial conditions between startup tests and TRACE data

Parameter	Startup tests	TRACE	
		Point kinetics	Difference (%)
Power (MWt)	2894	2894	0
Dome Pressure (MPa)	7.3	7.3	0
Feedwater Flow (kg/sec)	1647	1652	-0.3
Steam Flow (kg/sec)	1647	1652	-0.3
Core inlet flow (kg/sec)	10647	10521	1.2

3.1 Startup test-load rejection with bypass valves

Startup test-load rejection with bypass valves was performed in November 11, 1981 and the initial power was 2894 MWt. The purpose of this test was to confirm the functions of TCVs, BPV, SRVs and the response of system. Table 2 shows the sequences of startup test and TRACE. Their sequences are very similar. In this transient, when load rejection occurred, the TCV closed quickly. Then the BPV opened and reactor scrammed. When the water level reached level 3, the recirculation pumps were tripped. Finally, the BPV was reset at 6.48 MPa.

Figure 4~6 shows the results of startup test and TRACE. Fig. 4 depicts the power curves of startup test and TRACE. The trends of their curves are similar. The TCV fast closure tripped the reactor scram. Therefore, the power dropped after 0.3 sec. Fig. 5 compares the steam dome pressures of startup test and TRACE. The trends of the curves are approximately in agreement. The TCV closing caused the dome pressure to rise. Then, BPV and SRVs opened and led to the decline of dome pressure. Due to the dome pressure increase, it resulted in the core inlet flow rising during 0.5~2 sec (see Fig. 6). In other parameters comparisons (like steam flow, feedwater flow, and water level), their trends were also similar. Then, recirculation pumps trip caused the decrease of core inlet flow. In summary, the trends of TRACE prediction were consistent with startup test data but there were a few differences in the values of the prediction. Because we cannot find the detailed startup test data, we don’t know what the reasons cause the differences of TRACE results and startup test data.

Table 2 The comparison of sequences between startup test and TRACE data

Action (sec)	Startup test	TRACE
Transient started	0	0
TCV started to close	0.2	0.2
BPV started to open	0.21	0.21
Reactor scrammed	0.236	0.236
BPV fully opened	0.329	0.329
TCV fully closed	0.394	0.394
Water level reached level 3	2.2	3.1
Steam dome pressure peak	3.9 (7.43 MPa)	2.5 (7.36 MPa)
BPV reset at 6.48 MPa	16.3	18.4
End of analysis	—	20

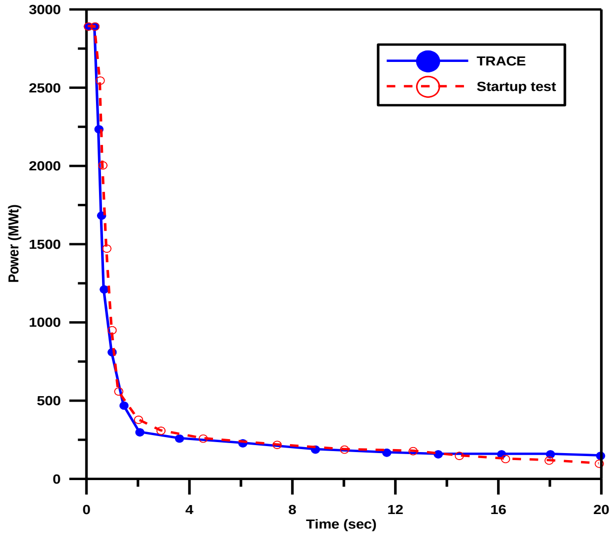


Fig. 4 Power data of TRACE and startup test.

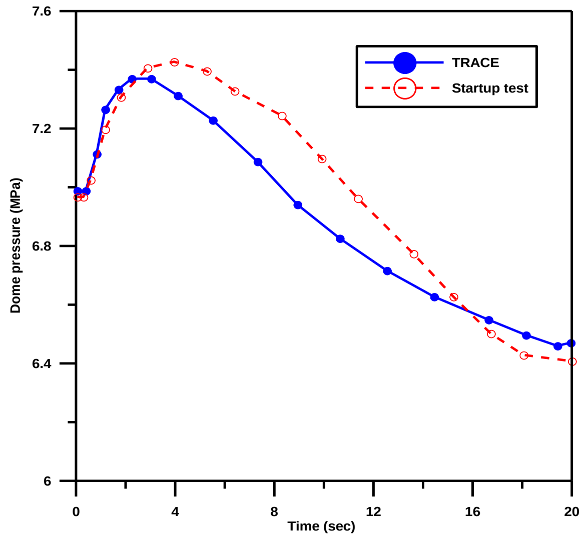


Fig. 5 Dome pressure data of TRACE and startup test.

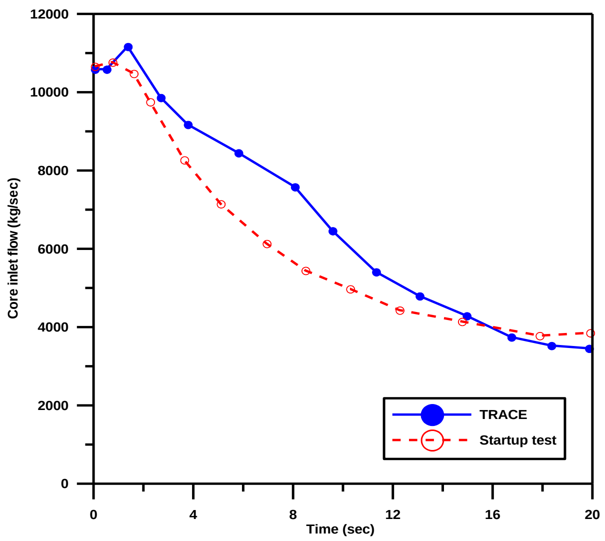


Fig. 6 Core inlet flow data of TRACE and startup test.

3.2 Startup Test-one feedwater pump trip

Startup Test-one feedwater pump trip was performed in November 6, 1981 and the initial power was 2778 MWt. The purpose of this test was to confirm the function of FCV (flow control valve) when one feedwater pump tripped. Table 3 compares the sequences of startup test with TRACE. Their sequences are nearly the same. In this transient, after one feedwater pump tripped, the water level decreased. When the water level reached level 4, the FCV runback was started. Then, the power and core flow decreased.

Figure 7~9 shows the results of startup test and TRACE. Fig. 7 depicts the core inlet flow curves of startup test and TRACE. The trends of their curves are similar. One feedwater pump trip caused the water level decrease. FCV runback was tripped when the water level reached level 4. Therefore, the core inlet flow decreased due to FCV runback. Fig. 8 compares the powers of startup test and TRACE. The trends of the curves are approximately in agreement. After FCV runback, the power decreased. Fig. 9 shows the feedwater flow data of startup test and TRACE. TRACE result was consistent with startup test data. One feedwater pump trip caused feedwater flow to decrease after 4.9 sec. Besides, the water level result of TRACE was similar to startup test data. In summary, the results of TRACE prediction were similar to startup test but there were a few differences in the values of parameters.

Table 3 The comparison of sequences between startup test and TRACE

Action (sec)	Startup test	TRACE
Transient Started	0	0
One feedwater pump tripped	4.9	4.9
Water level dropped to level 4	15.1	15.4
Minimum power value	18.5 (57%)	18.5 (57%)
Minimum core flow	19.4 (79.3%)	18.7 (77%)
End of analysis	—	30

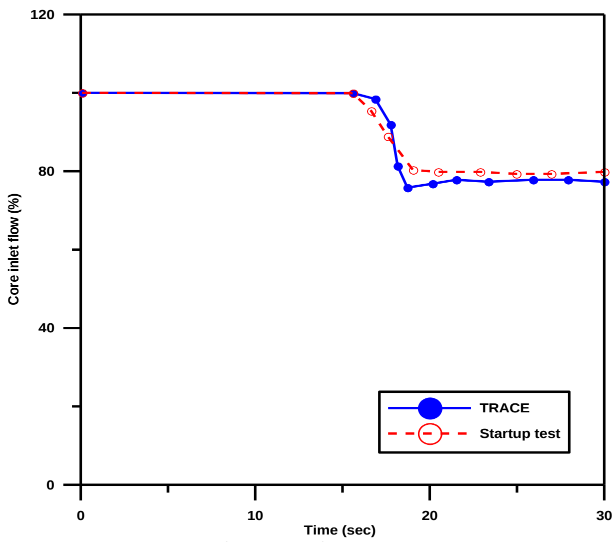


Fig. 7 Core inlet flow data of TRACE and startup test.

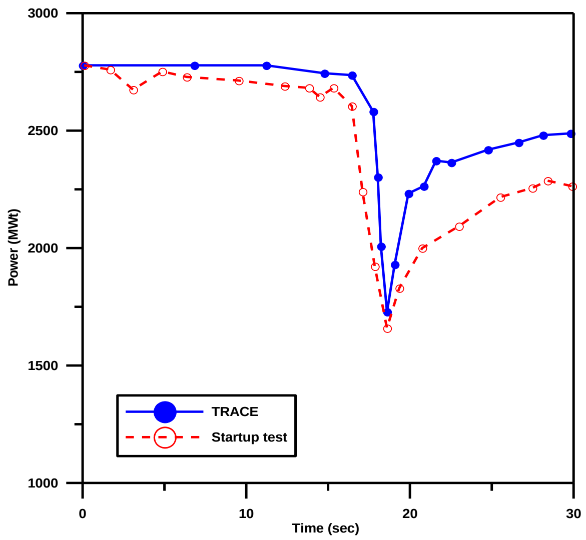


Fig. 8 Power data of TRACE and startup test.

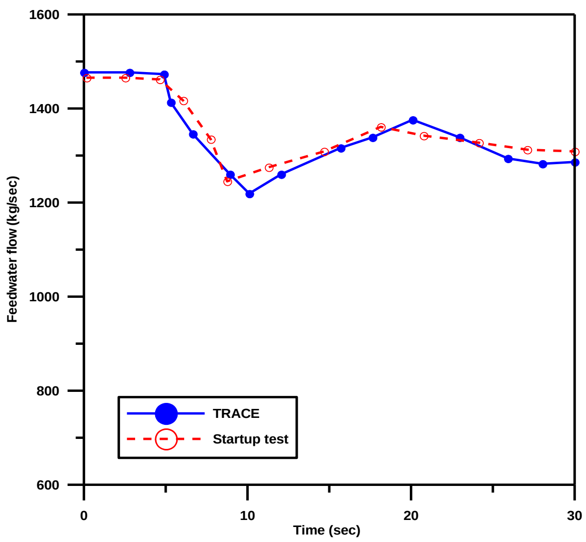


Fig. 9 Feedwater flow data of TRACE and startup test.

By the above TRACE and startup tests comparisons, it indicates that there is a respectable accuracy in Kuosheng NPP TRACE/SNAP model.

3.3 SBO + LOCA transient analysis (no water injection case)

Besides, in order to estimate the safety of Kuosheng NPP under the more severe conditions, the SBO + LOCA transient analysis of Kuosheng NPP TRACE/SNAP model was performed. There were two cases in the SBO + LOCA transient analysis, as follows: 1. No water injection case, 2. fire water injection case (RCIC assumed failure). Besides, there were some assumptions in this transient, including: (1) the simulation of steady state was performed during 0~200 sec; (2) the scram of reactor, all recirculation pumps trip, feedwater flow trip, MSIVs closed were performed at 200 sec; (3) LOCA occurred in one steamline (before MSIV) at 200 sec; (4) the decay heat model ANS-73 was used in this transient.

In no water injection case, it assumed that no water injected into the vessel during the SBO + LOCA transient. When LOCA occurred, the pressure of drywell and wetwell increased due to the large amount of steam was released into the drywell (see Fig. 12). Then, the large amount of steam also caused that the temperature of drywell went up (see Fig. 10). According to FSAR [10], the limit of drywell temperature is 438.71 K. But the drywell temperature in this case reached the limit at 1680 sec. It indicated that there was a safety issue in drywell. Besides, the water amount of the vessel still decreased after LOCA occurred because no water injected to the vessel (shown in Fig. 11). The core water level was lower than TAF at about 700 sec. It made PCT increase after 700 sec. The PCT was larger than 1088.7 K at about 1900 sec which resulted in the zirconium-water reaction generation. This result indicated that the integrity of fuel rod was not able to keep.

Figure 13~16 show the analysis results of FRAPTRAN. The zirconium-water reaction occurred at 1900 sec (see Fig. 13). According to 10 CFR 50.46 rule [14], the increasing oxide thickness of cladding should be less than 17%. But the cladding oxide thickness in this case was over the critical value (see Fig. 14). Besides, FRAPTRAN output file also depicted the fuel rod burst in the node 18. It indicated that the integrity of fuel rod was not kept. Fig. 15 and 16 illustrate the cladding hoop strain and stress results of FRAPTRAN. When the cladding temperature increased, the cladding hoop strain and stress also rose. NUREG-0800 Standard Review Plan [15] clearly defines fuel cladding failure criteria. For the uniform strain value, it is limited not to exceed 1%. The cladding hoop strain (node 18) went up sharply after 2000 sec and was larger than this limit. The cladding hoop stress drops abruptly to zero after 2000 sec. The above results also indicated that the integrity of cladding did not keep after 2000 sec.

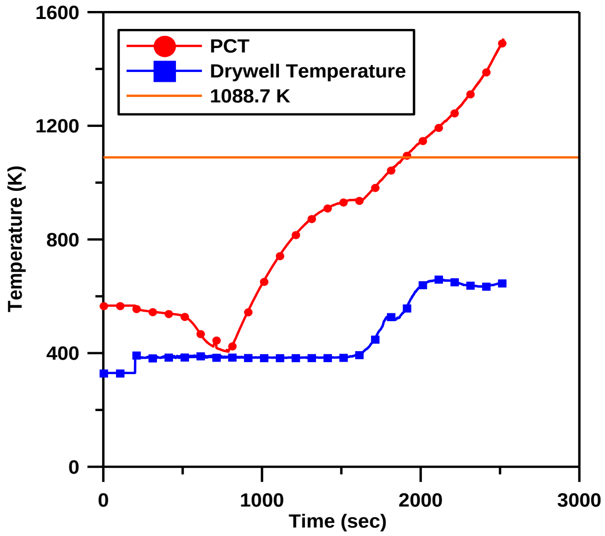


Fig. 10 PCT and drywell temperature results of TRACE.

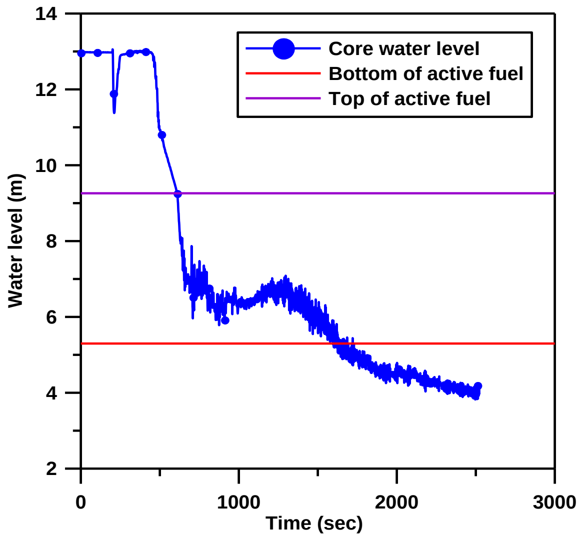


Fig. 11 Core water level results of TRACE.

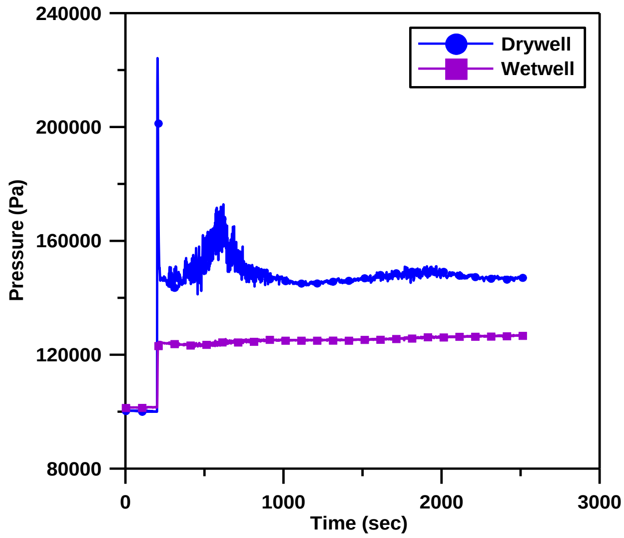


Fig. 12 Drywell and wetwell pressure results of TRACE.

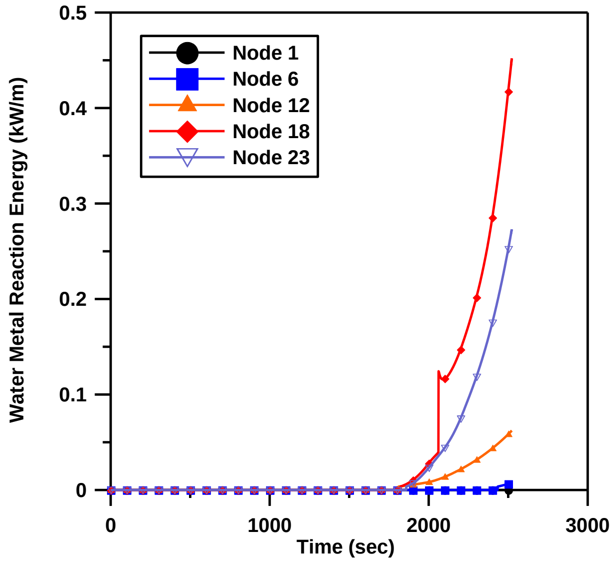


Fig. 13 Water metal reaction energy results of FRAPTRAN.

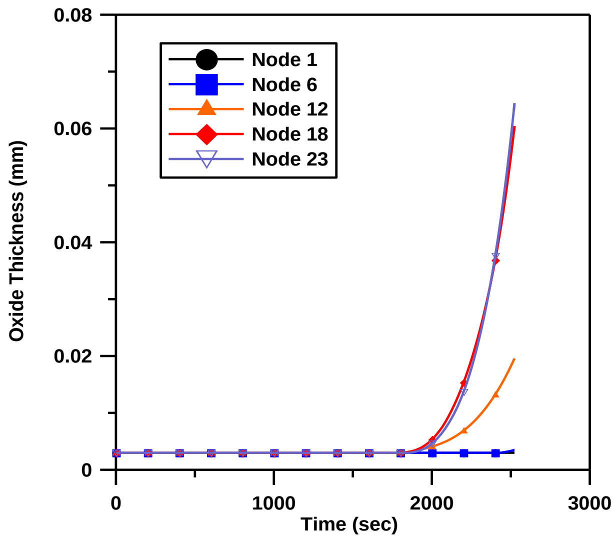


Fig. 14 Oxide thickness results of FRAPTRAN.

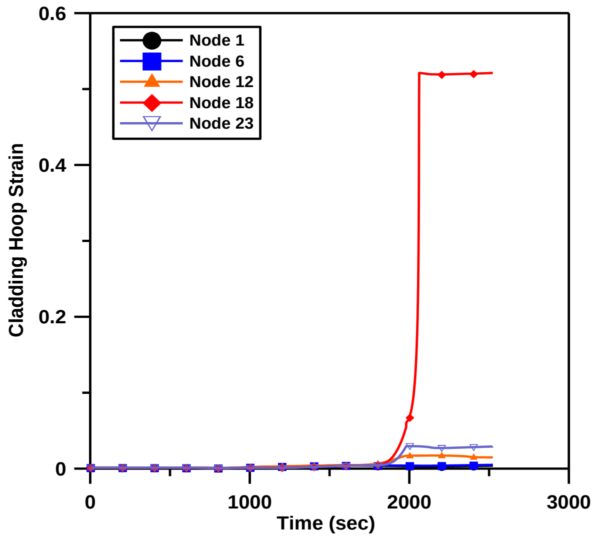


Fig. 15 Cladding hoop strain results of FRAPTRAN.

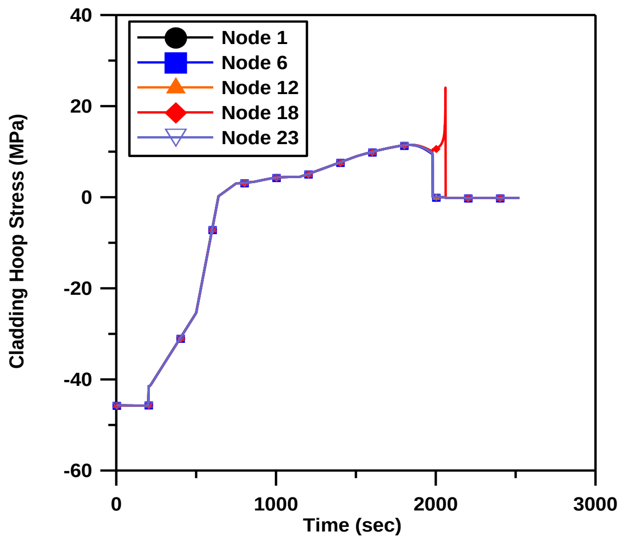


Fig. 16 Cladding hoop stress results of FRAPTRAN.

3.4 SBO + LOCA transient analysis (fire water injection case)

In fire water injection case, it assumed that fire water (flow rate assumed 39 kg/sec) started to inject into the vessel after 800 sec. RCIC was assumed to fail in this case. Fig. 17~19 illustrate the analysis results of TRACE/SNAP. When LOCA occurred, the pressure of drywell and wetwell increased due to the large amount of steam was released into the drywell (shown in Fig. 19). Then, the large amount of steam also caused that the temperature of drywell went up (see Fig. 17). However, drywell temperature in this case reached the limit (438.71 K) at 1460 sec and was lower than the limit at 2570 sec. It indicated that there was a safety issue in drywell during 1460~2570 sec. Besides, the water amount of the vessel decreased after LOCA occurred and increased after 1000 sec (shown in Fig. 18). The core water level was lower than TAF at about 700 sec and PCT started to increase at 800 sec. However, because fire water injected into the vessel, the PCT started to decrease after 2200 sec which did not reach 1088.7 K. It indicated that the integrity of fuel rod was kept. Fig. 20 and 21 show the cladding hoop strain and stress results of FRAPTRAN. When the cladding temperature increased, the cladding hoop strain and stress also went up. The cladding hoop strain and stress decreased as the cladding temperature dropped.

Finally, by the animation function of SNAP with TRACE analysis results, the animation of the no water injection case was presented in Fig. 2 and 22. This transient started at 200 sec (see Fig. 2) and the results of TRACE/SNAP in this transient can be observed in Fig. 22.

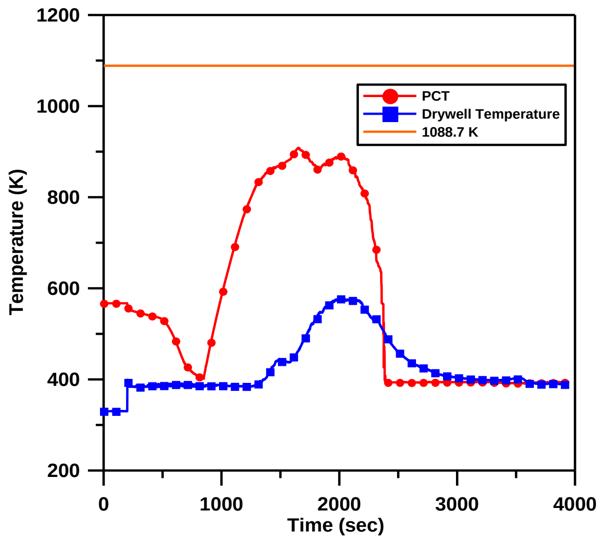


Fig. 17 PCT and drywell temperature results of TRACE.

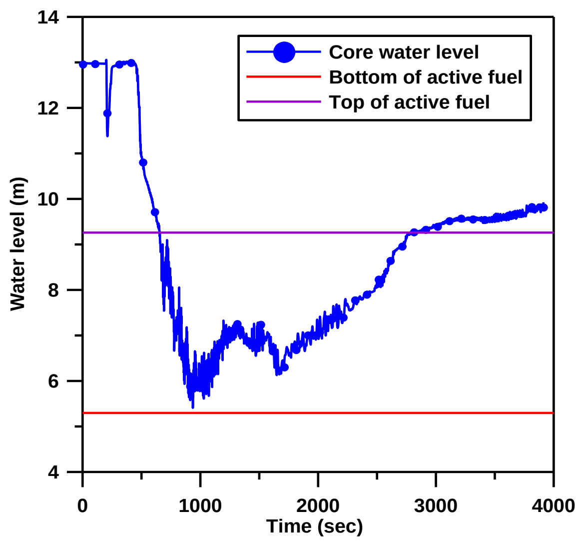


Fig. 18 Core water level results of TRACE.

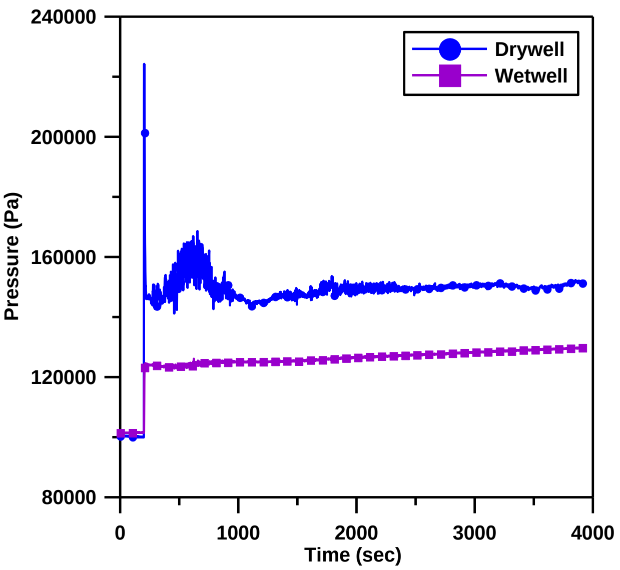


Fig. 19 Drywell and wetwell pressure results of TRACE.

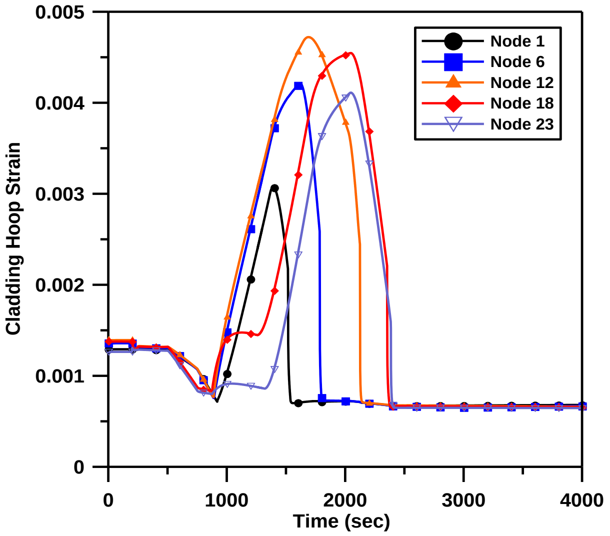


Fig. 20 Cladding hoop strain results of FRAPTRAN.

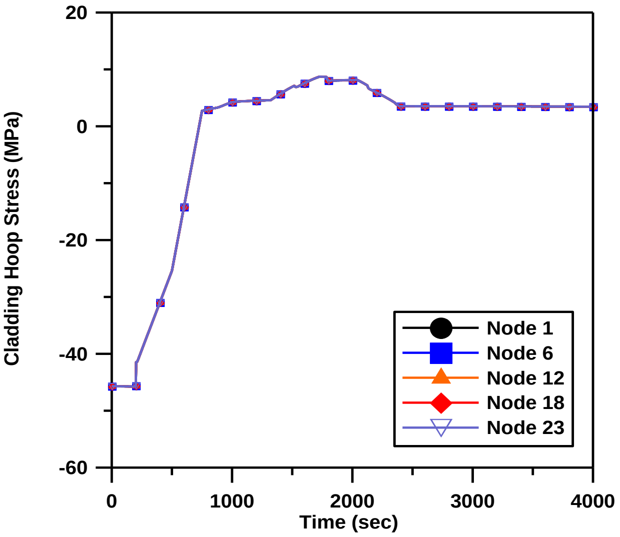
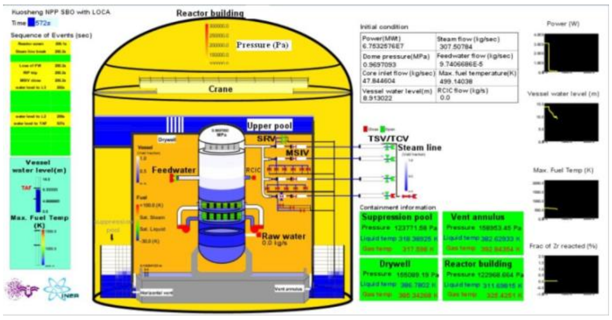
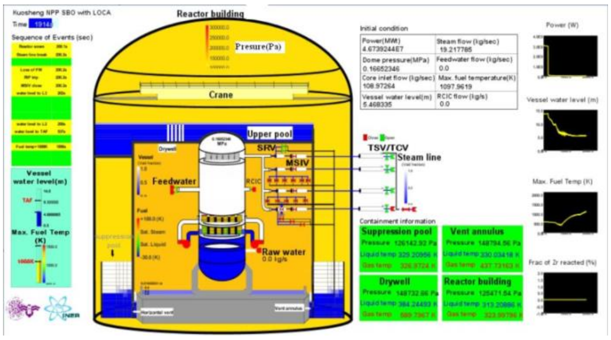


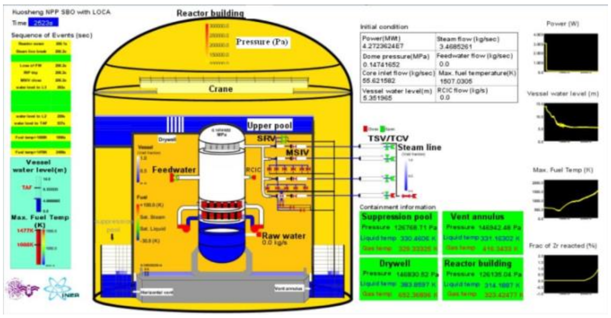
Fig. 21 Cladding hoop stress results of FRAPTRAN



(a) at 572 sec



(b) at 1914 sec



(c) at 2523 sec

Fig. 22 SNAP animation model of Kuosheng NPP for the no water injection case (a) at 572 sec (b) at 1914 sec (c) at 2523 sec.

4 Conclusion

This research focuses on the establishment of Kuosheng NPP TRACE/SNAP model. The load rejection and a feedwater pump trip transients were selected to assess Kuosheng NPP TRACE/SNAP model. The results and sequences of TRACE were similar to startup tests data. By the above compared results, it indicates that there is a respectable accuracy in Kuosheng NPP TRACE/SNAP model and it also shows that Kuosheng NPP TRACE/SNAP model is satisfying for the purpose of Kuosheng NPP safety analyses with confidence.

In SBO + LOCA transient (no water injection case), PCT reached the criteria of 1088.7 K at 1900 sec and FRAPTRAN results indicated that the fuel rod burst at 2000 sec. The cladding oxide thickness in this case was over the critical value (17%) and cladding hoop strain (node 18) was larger than the limit (1%). Besides, the drywell temperature reached the limit (438.71 K) at 1680 sec. It indicated that there was a safety issue in drywell after 1680 sec.

However, if the fire water (flow rate 39 kg/sec) injected to the vessel at 800 sec in this transient, TRACE results depicted that PCT was lower than 1088.7 K and FRAPTRAN results also indicated that the integrity of fuel rod was kept. But one safety issue generated in the drywell during 1460~2570 sec due to the drywell temperature larger than the limit (438.71 K). Finally, TRACE/SNAP analysis results were presented by the animation model of Kuosheng NPP.

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