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International Agreement Report

The Development and Application of Kuosheng (BWR/6) Nuclear Power Plant TRACE/SNAP Model

Prepared by:
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ABSTRACT

Kuosheng Nuclear Power Plant (NPP) is the second NPP in Taiwan which is the BWR/6 plant. This research focuses on the development of the Kuosheng NPP TRACE/SNAP model. In order to check the system response of the Kuosheng NPP TRACE/SNAP model, this study uses the analysis results of Final Safety Analysis Report (FSAR) and startup tests data to assess the Kuosheng NPP TRACE/SNAP model. The load rejection and a feedwater pump trip transients were selected to validate the Kuosheng NPP TRACE/SNAP model. The trends of TRACE analysis results were consistent with the FSAR and startup tests data. It indicates that there is a respectable accuracy in the Kuosheng NPP TRACE/SNAP model. Besides, this research also focuses on the application of the Kuosheng NPP TRACE/SNAP model in the core shroud leakage. The core shroud leakage is one of issues of concern by the U.S. NRC and NPPs. This research is using the Kuosheng NPP TRACE/SNAP model to perform the core shroud leakage transients for Kuosheng NPP safety analysis. The TRACE analysis results show that the pure core shroud leakage transient wasn't influence the Kuosheng NPP safety. However, the core shroud leakage + station blackout (SBO) / steamline break (loss of coolant accident, LOCA) transient caused the cladding temperature larger than 1088 K and affected the Kuosheng NPP safety.

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. INER (Institute of Nuclear Energy Research, Atomic Energy Council, R.O.C.) is the organization in Taiwan responsible for the application of TRACE in thermal hydraulic safety analysis, for recording user's experiences of it, and providing suggestions for its development. To meet this responsibility, the TRACE/SNAP model of Kuosheng NPP has been built. In this report, this study uses FSAR and startup tests data to assess the Kuosheng NPP TRACE/SNAP model.

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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/SNAP model of Kuosheng 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. TRACE also can provide greater simulation capability than the previous codes, especially for events like LOCA.

Kuosheng NPP is located on the northern coast of Taiwan. Its nuclear steam supply system is a type of BWR/6 designed and built by General Electric on a twin unit concept. Each unit includes two loops of recirculation piping and four main steam lines, with the thermal rated power of 2894MWt. After the project of MURPU (Measurement Uncertainty Recovery Power Uprate) for Kuosheng NPP, the operating power is 101.7% of the original designed rated power, which is 2943 MWt now. Unit 1 will start SPU (Stretch Power Uprate) from Cycle 24 and Unit 2 will start SPU from Cycle 23. The operating power will be 104.7% of the OLTP (Original Licensed Thermal Power), which will be 3030 MWt. In order to check the system response of the Kuosheng NPP TRACE/SNAP model, this study uses the analysis results of FSAR and startup tests data to assess the Kuosheng NPP TRACE/SNAP model. The load rejection and a feedwater pump trip transients were selected to validate the Kuosheng NPP TRACE/SNAP model. The trends of TRACE analysis results were consistent with the FSAR and startup tests data. It indicates that there is a respectable accuracy in the Kuosheng NPP TRACE/SNAP model. Besides, this research also focuses on the application of the Kuosheng NPP TRACE/SNAP model in the core shroud leakage. The core shroud leakage is one of issues of concern by the U.S. NRC and NPPs. This research is using the Kuosheng NPP TRACE/SNAP model to perform the core shroud leakage transients for Kuosheng NPP safety analysis. The TRACE analysis results show that the pure core shroud leakage transient wasn't influence the Kuosheng NPP safety. However, the core shroud leakage + SBO/LOCA transient caused the cladding temperature larger than 1088 K and affected the Kuosheng NPP safety.

ABBREVIATIONS

BPV	Bypass Valve
BWR	Boiling Water Reactor
CAMP	Code Applications and Maintenance Program
FSAR	Final Safety Analysis Report
IGSCC	Intergranular Stress Corrosion Cracking
INER	Institute of Nuclear Energy Research Atomic Energy Council, R.O.C.
LOCA	Loss of Coolant Accident
MSIV	Main Steamline Isolation Valve
MURPU	Measurement Uncertainty Recovery Power Uprate
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRWL	Narrow Range Water Level
RPV	Reactor Pressure Vessel
OLTP	Original Licensed Thermal Power
SBO	Station Blackout
SNAP	Symbolic Nuclear Analysis Package
SRV	Safety Relief Valve
TCV	Turbine Control Valve
TPC	Taiwan Power Company
TRACE	TRAC/RELAP Advanced Computational Engine
TSV	Turbine Stop Valve

1. INTRODUCTION

Kuosheng NPP is located on the northern coast of Taiwan. Its nuclear steam supply system is a type of BWR/6 designed and built by General Electric on a twin unit concept. Each unit includes two loops of recirculation piping and four main steam lines, with the thermal rated power of 2894 MWt. After the project of MURPU for Kuosheng NPP, the operating power is 101.7% of the original designed rated power, which is 2943 MWt now. Besides, Unit 1 will start SPU from Cycle 24 and Unit 2 will start SPU from Cycle 23. The operating power will be 104.7% of the OLTP, which will be 3030 MWt. The safety analysis of the NPP is very important work in the NPP safety. The importance of NPP safety analysis has been raised and there is more concern for the safety of the NPPs in the world after the Fukushima NPP event occurred. The advanced thermal hydraulic code named TRACE has been developed by U.S. NRC for NPP safety analysis. According to the user manual [1], 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. The development of TRACE is based on TRAC, combining with the capabilities of RELAP5 and other programs. SNAP, a graphic user interface program that processes the inputs and outputs of TRACE, has also been developing. 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 offers the greater simulation capability than other old codes, especially for events such as LOCA.

This research focuses on the development of the Kuosheng NPP TRACE/SNAP model. In our previous study [2], the preliminary TRACE/SNAP model of Kuosheng NPP was established. The Kuosheng NPP TRACE/SNAP model includes one 3-D vessel, six channels which are used to simulate 624 fuel bundles, four steamlines, and 16 SRVs components, etc.. In order to check the system response of the Kuosheng NPP TRACE/SNAP model, this study uses the FSAR and startup tests data to assess the Kuosheng NPP TRACE/SNAP model. The load rejection and a feedwater pump trip transients were selected to validate the Kuosheng NPP TRACE/SNAP model.

Besides, this research also focuses on the application of the Kuosheng NPP TRACE/SNAP model in the core shroud leakage. Intergranular stress corrosion cracking (IGSCC) of BWR internal components has been identified as a technical issue of concern by the U.S. NRC and NPPs. The core shroud is among the list of internals susceptible to IGSCC. So IGSCC may cause core shroud leakage generated. Therefore, the core shroud leakage safety analysis of Kuosheng NPP was performed by the Kuosheng NPP TRACE/SNAP model in this study. Besides, refer to the conditions of Fukushima NPP event, the core shroud leakage + SBO/LOCA transients were also analyzed by TRACE for Kuosheng NPP.

2. MODEL OF KUOSHENG (BWR/6) NPP

The code versions adopted in this research are SNAP v 2.2.7 and TRACE v 5.0p3. The process of Kuosheng NPP TRACE/SNAP model development is as follows (shown in Fig. 1): First, the system and operating data for the cases of FSAR and startup tests of Kuosheng NPP are collected [1]-[5]. Second, several important control systems such as recirculation flow control system, pressure control system and feedwater flow control system etc. are established by SNAP and TRACE. Next, other necessary components (e.g., RPV (Reactor pressure vessel) and main steam piping) are added into the TRACE/SNAP model to complete the TRACE/SNAP model for Kuosheng NPP. Finally, the Kuosheng TRACE/SNAP model is verified with the cases of FSAR and startup tests. The TRACE/SNAP model of Kuosheng NPP is presented in Fig. 2. SNAP also can use the TRACE results data to make an animation for transient, such as Fig. 3.

Kuosheng NPP TRACE/SNAP model has been built according to the FSAR, startup tests data, design documents, and TRACE manuals [1]-[5]. Kuosheng NPP reactor, divided into two azimuthal sectors, four radial rings, eleven axial levels, altogether eighty-eight cells, are simulated by the 3-D vessel component (Fig. 4). Six channels, which are one dimensional component, are used for simulating 624 fuel bundles (Fig. 5). Full length fuel rods, partial length fuel rods and water rods are also simulated in the channel components. Two recirculation loops are set outside the reactor, with a recirculation pump in each loop (Fig. 6). 10 groups of jet pumps are merged into an equal jet pump (Fig. 7). Four steam lines connected with the vessel and each steam line had one MSIV (main steamline 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 use valve components to simulate the MSIV, SRVs, TCV, TSV and BPV (Fig. 8). The critical flow models for the MSIVs, SRVs, TCVs, TSVs, and BPV have been considered in our analysis. The break components were used to simulate the boundary conditions of turbine and condenser (Fig. 9). 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. 2 and 10~13. In the Kuosheng NPP TRACE/SNAP model, there are three simulation control systems included (1) feedwater 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 are provided as TRACE input for power calculations (Fig. 14). The initial conditions of Kuosheng NPP were also presented in Fig. 3.

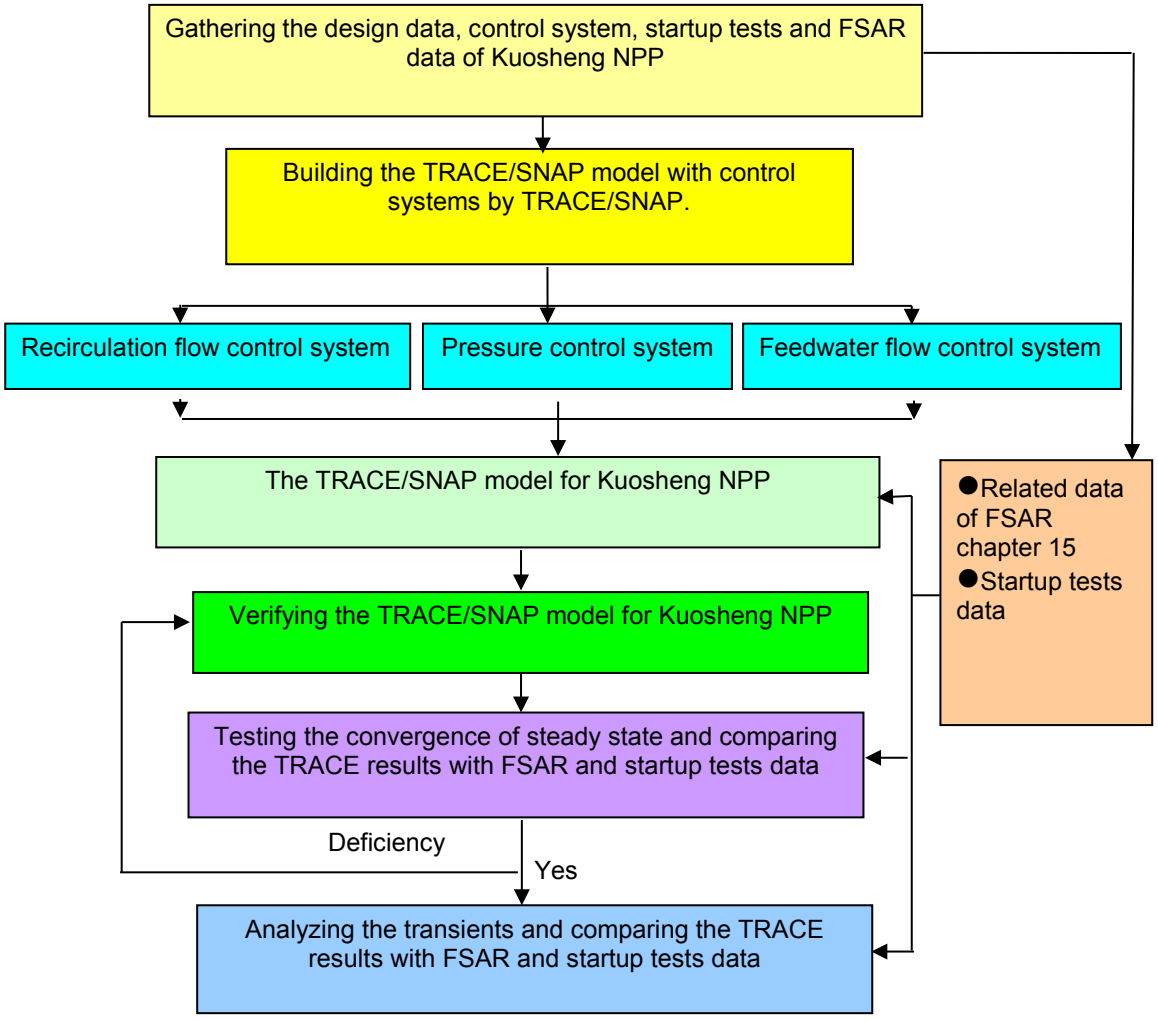


Figure 1 The flow chart of establishing and verifying the TRACE/SNAP model of Kuosheng NPP

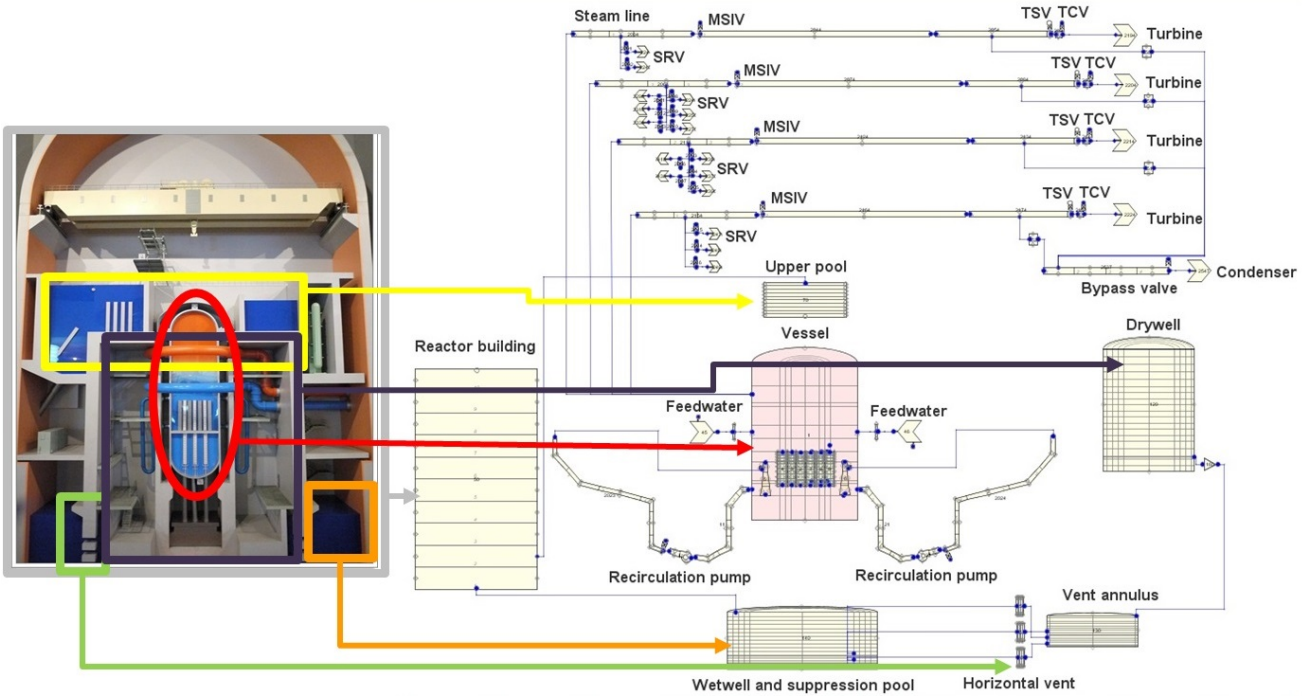


Figure 2 The TRACE/SNAP model of Kuosheng NPP

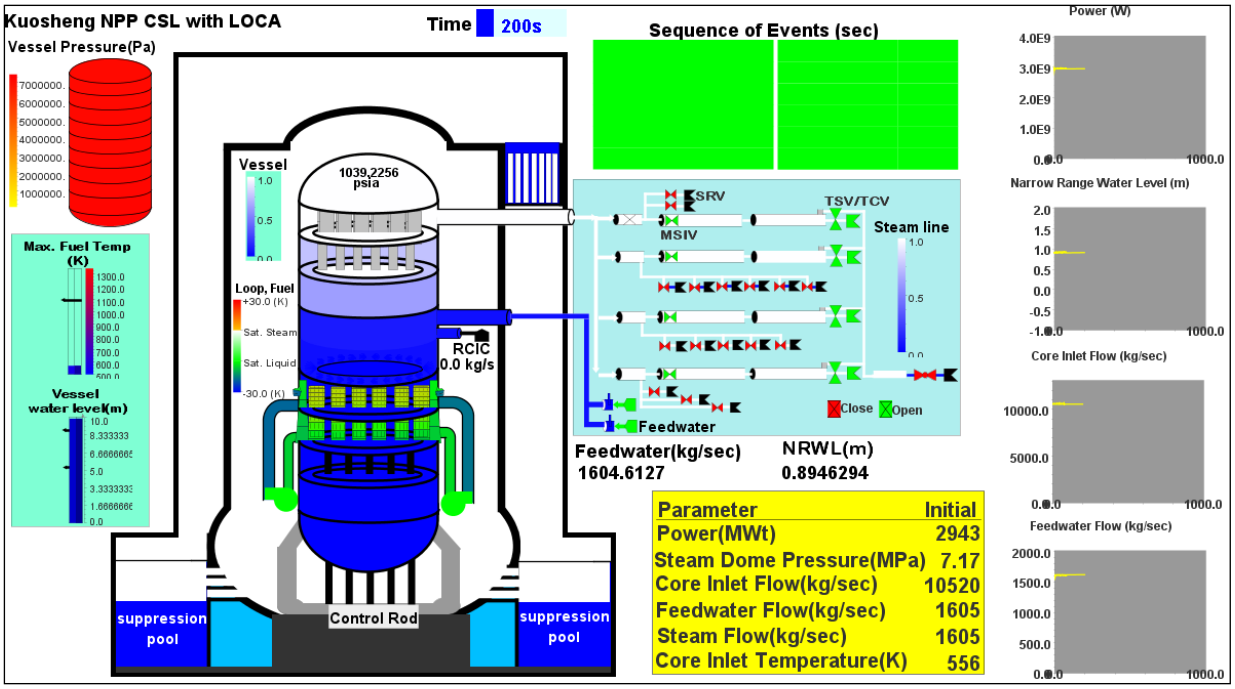


Figure 3 The animation model of Kuosheng NPP

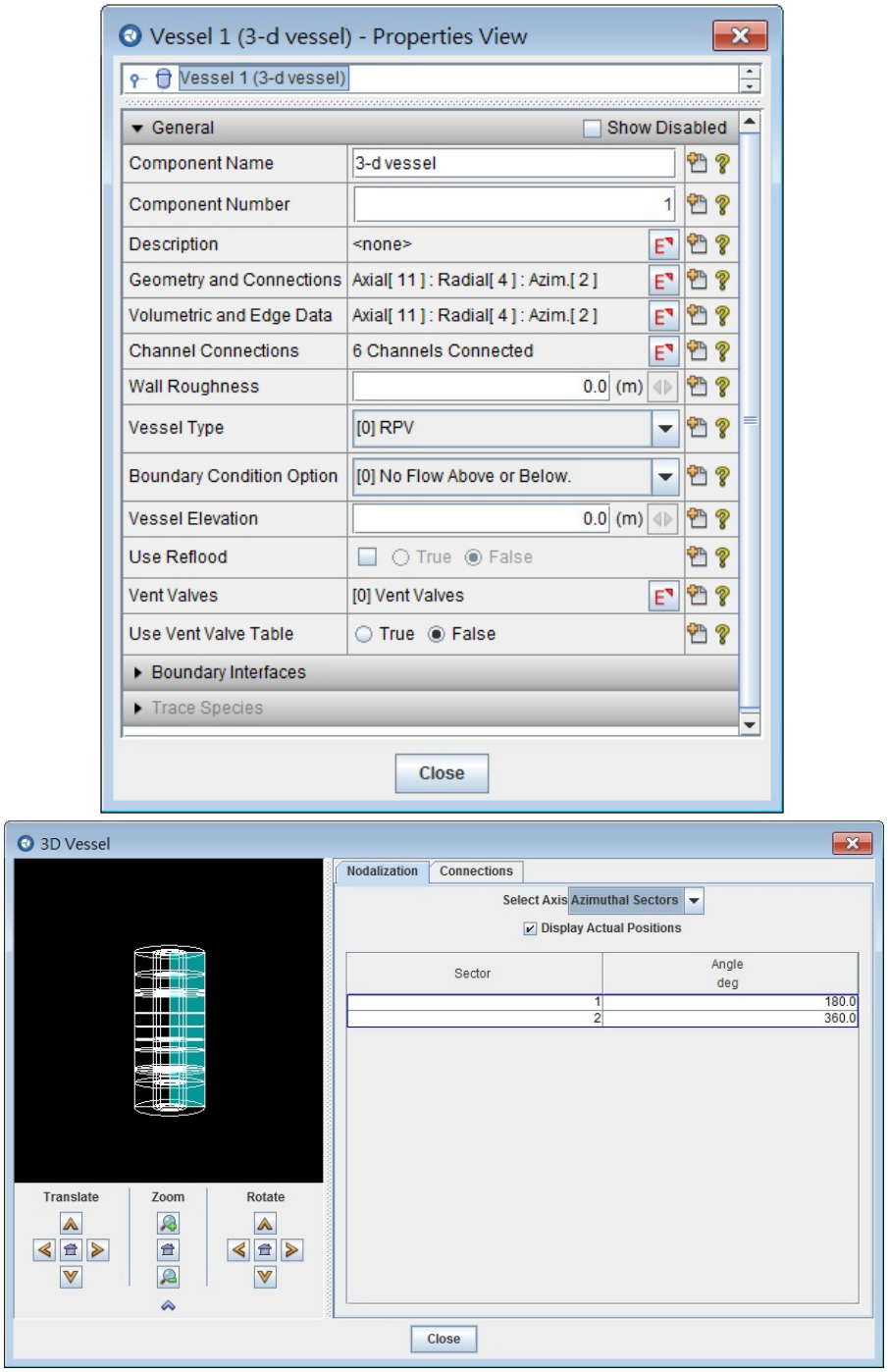


Figure 4 The 3D vessel condition of Kuosheng NPP

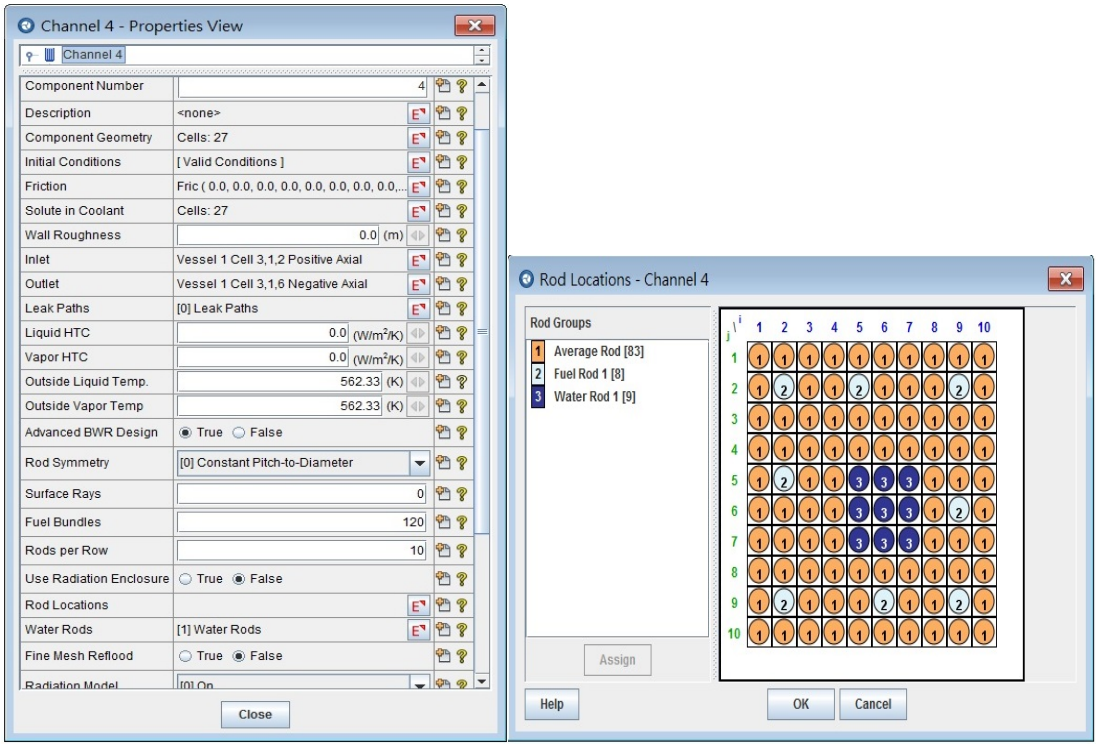


Figure 5 The channel condition of Kuosheng NPP

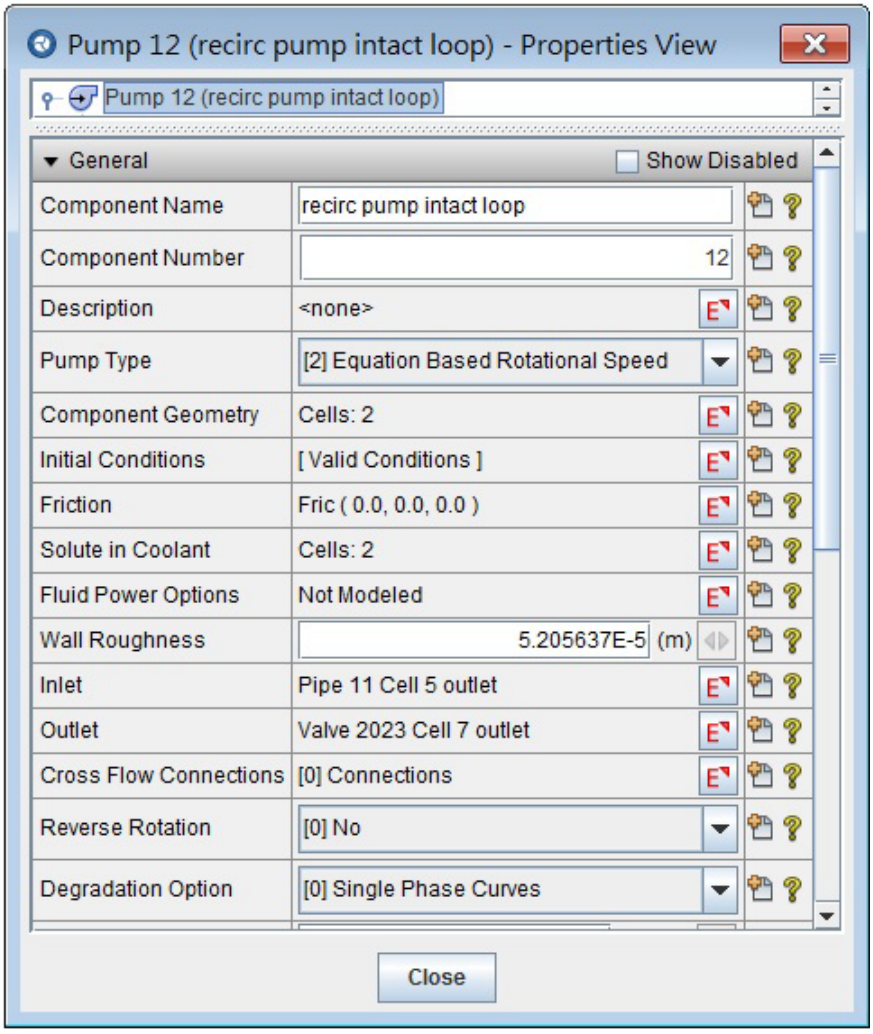


Figure 6 The recirculation pump condition of Kuosheng NPP

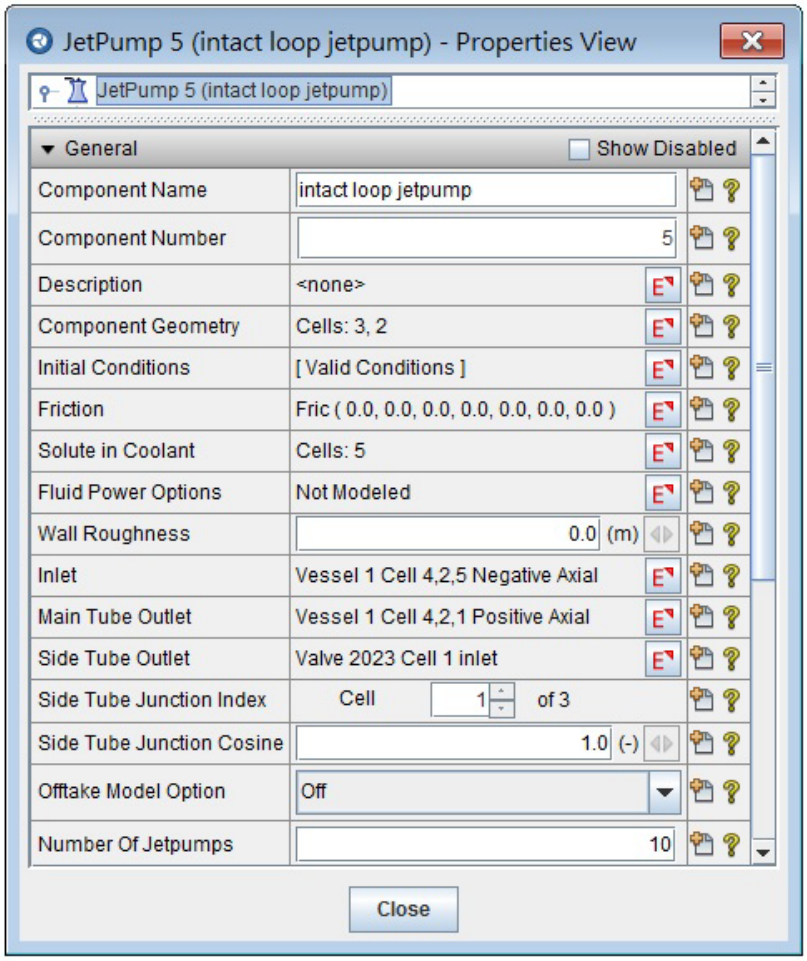


Figure 7 The jetpump condition of Kuosheng NPP

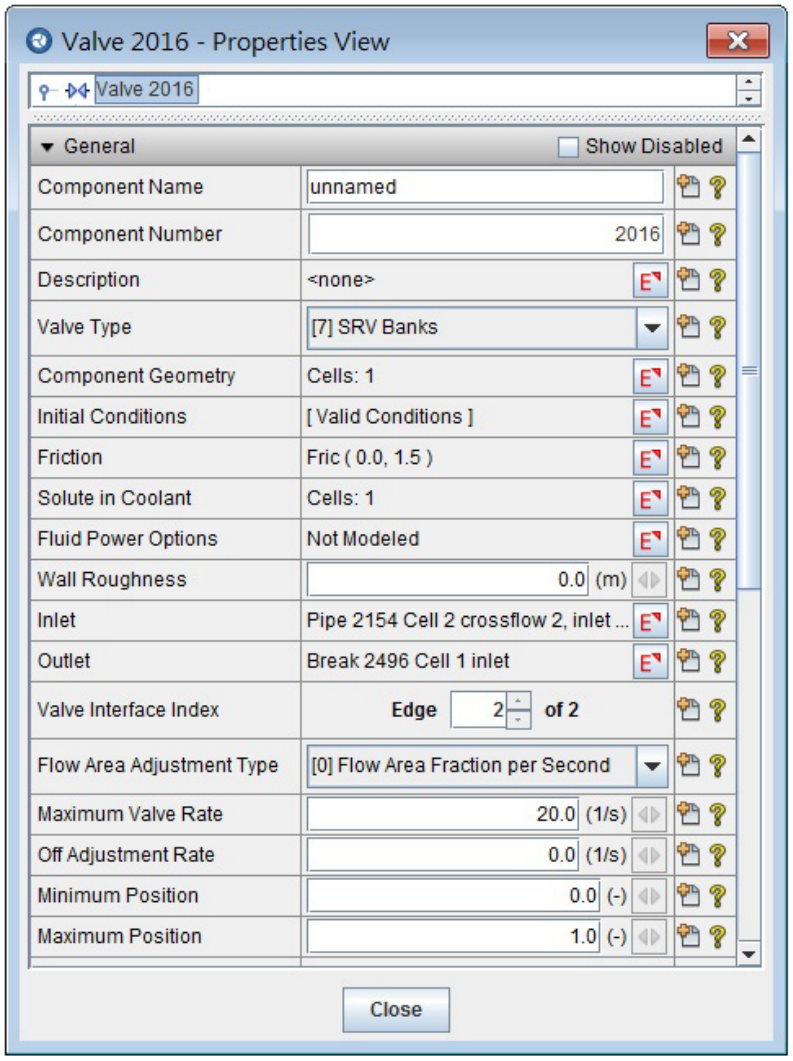


Figure 8　The SRV condition of Kuosheng NPP

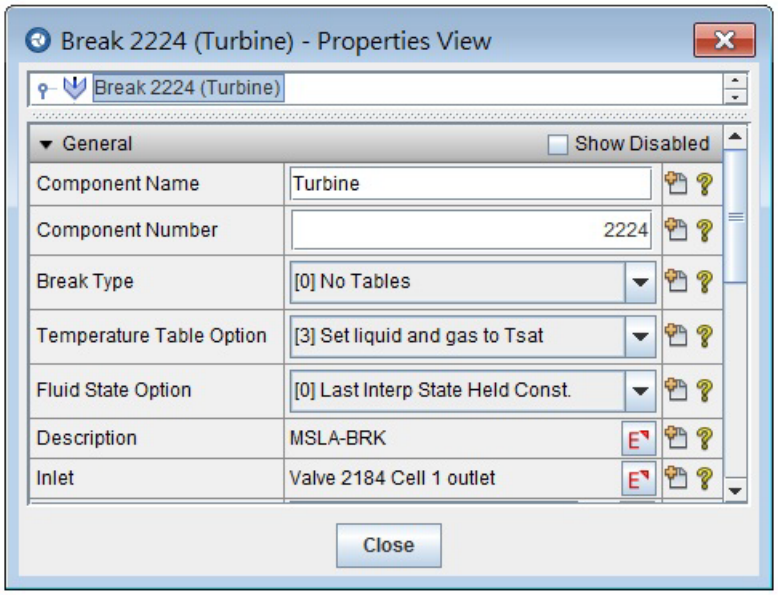


Figure 9 The turbine condition of Kuosheng NPP

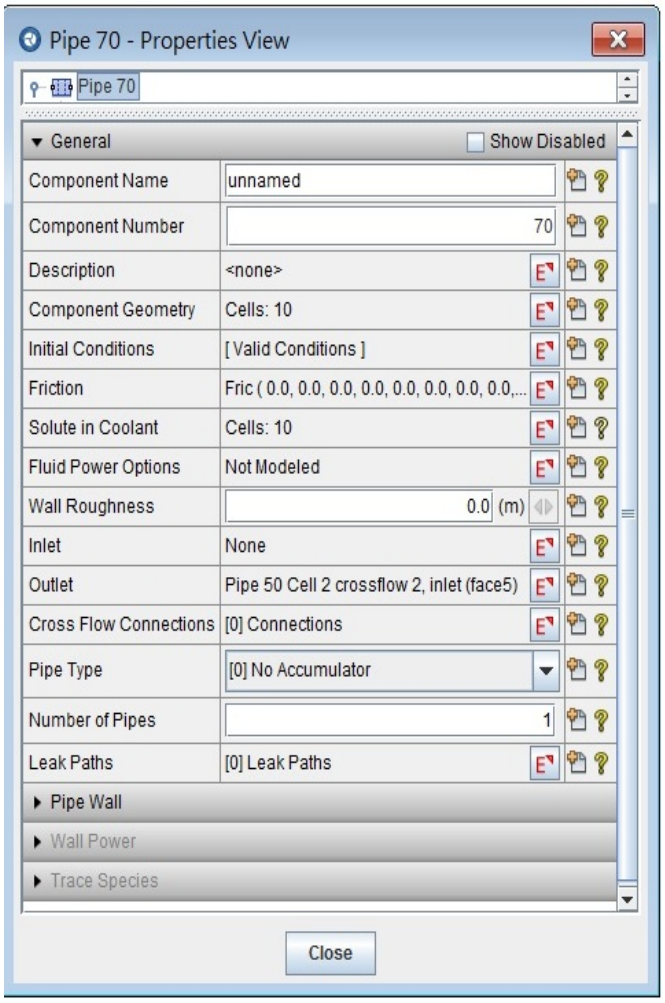


Figure 10 The upper pool condition of Kuosheng NPP

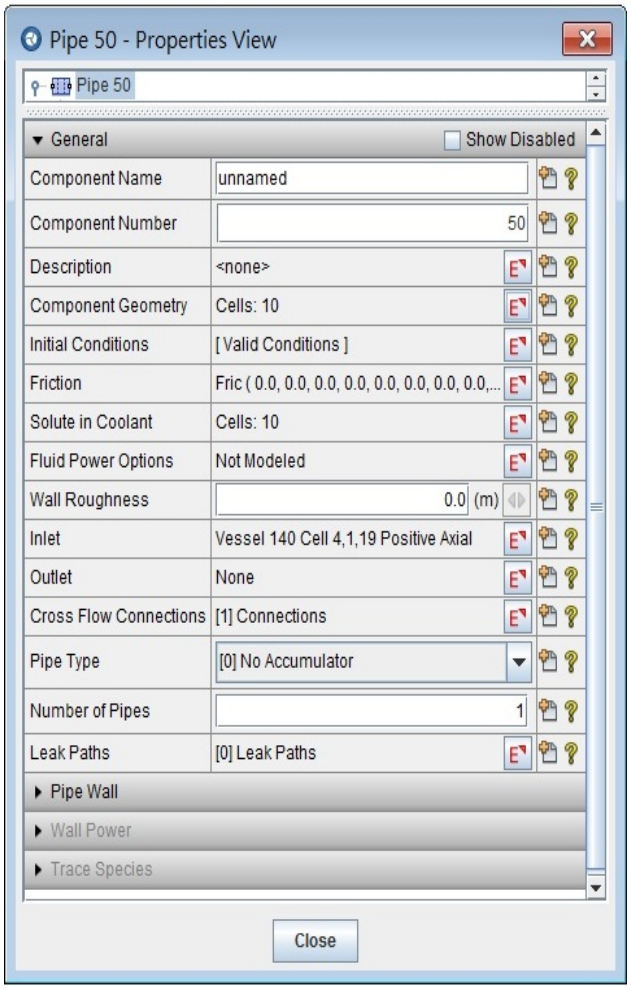


Figure 11 The reactor building condition of Kuosheng NPP

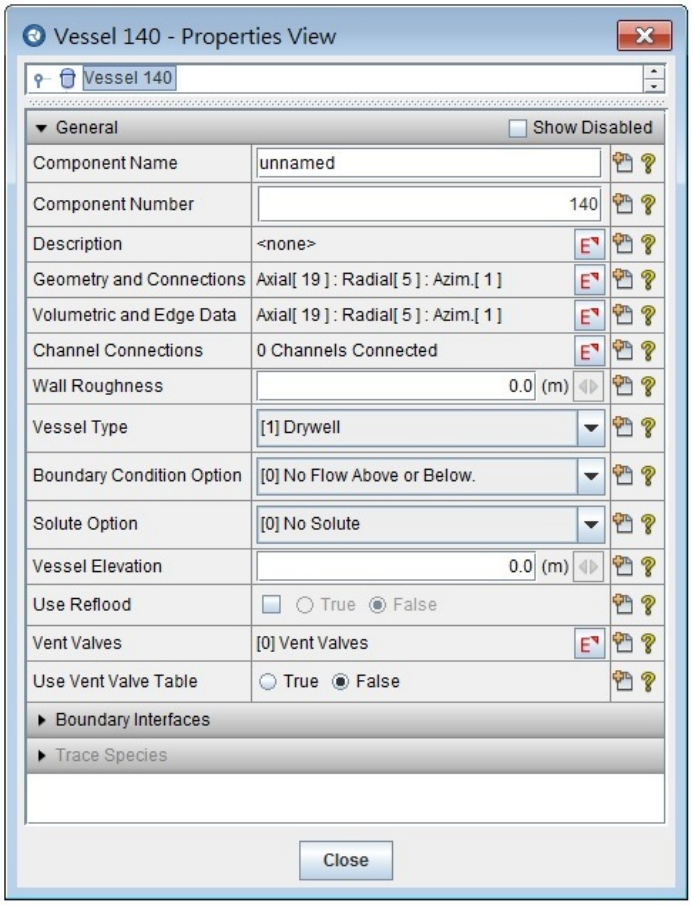


Figure 12 The wetwell and suppression pool condition of Kuosheng NPP

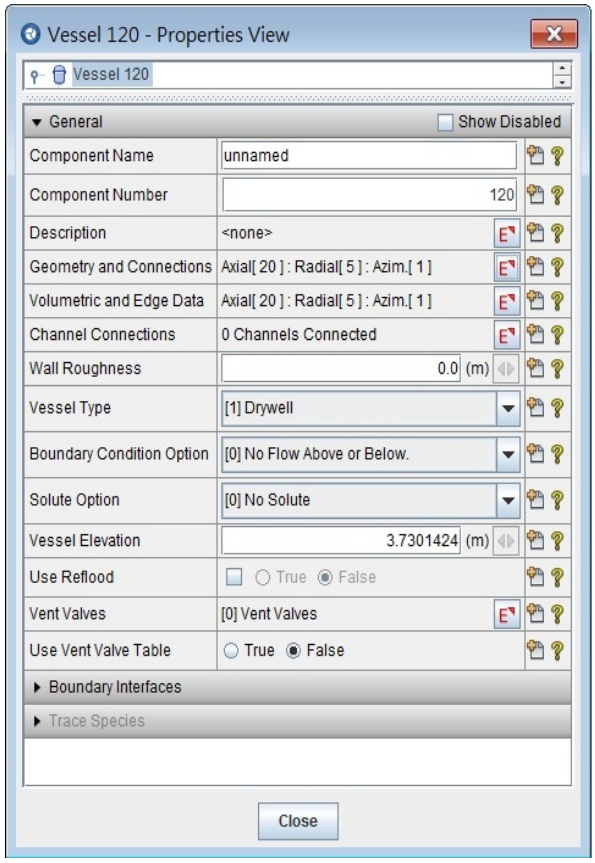


Figure 13 The drywell condition of Kuosheng NPP

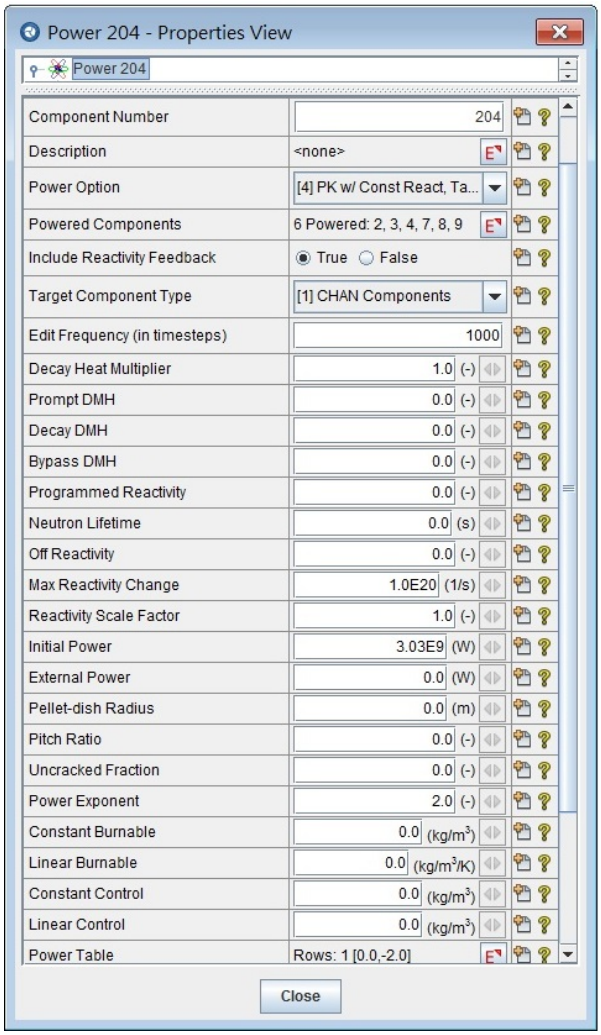


Figure 14 The power condition of Kuosheng NPP

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 FSAR data under the steady state condition. The results of analysis of TRACE are clearly consistent with FSAR data under the steady state condition (See Table 1).

Table 1 The comparison of initial conditions between FSAR and TRACE data

Parameter	FSAR	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 : Generator Load Rejection with Bypass Valves

This test was performed in November 11, 1981 and the initial power was 2894 MWt. The purpose of the test was to confirm the functions of TCVs, BPV, SRVs and the response of system. Table 2 compares the generator load rejection with bypass valves transient's sequences of startup test data with 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.

Fig.15~20 shows the results of startup test and TRACE. Fig. 15 depicts the power curves of startup test data 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. 16 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. Fig. 17 shows the steam flow data of startup test and TRACE. The TRACE result was consistent with the startup test data. The TCV closing caused the steam flow to drop sharply after 0.4 sec. Besides, the feedwater flow and water level decreased after TCV closure (shown in Fig. 18 and 19). Due to the dome pressure increase, it resulted in the core inlet flow rising during 0.5~2 sec (see Fig. 20). 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

Event(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

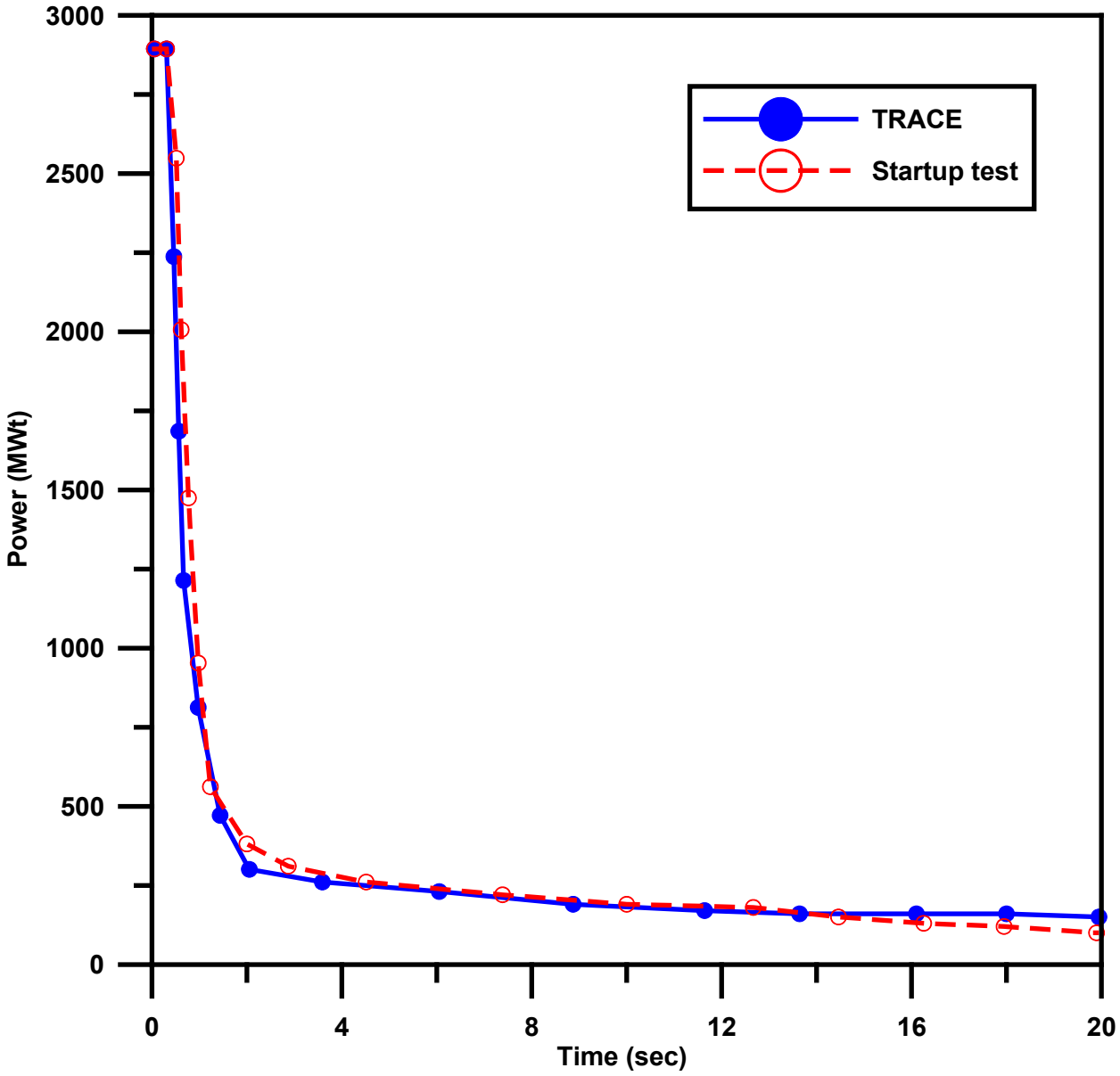


Figure 15 The power data of TRACE and startup test

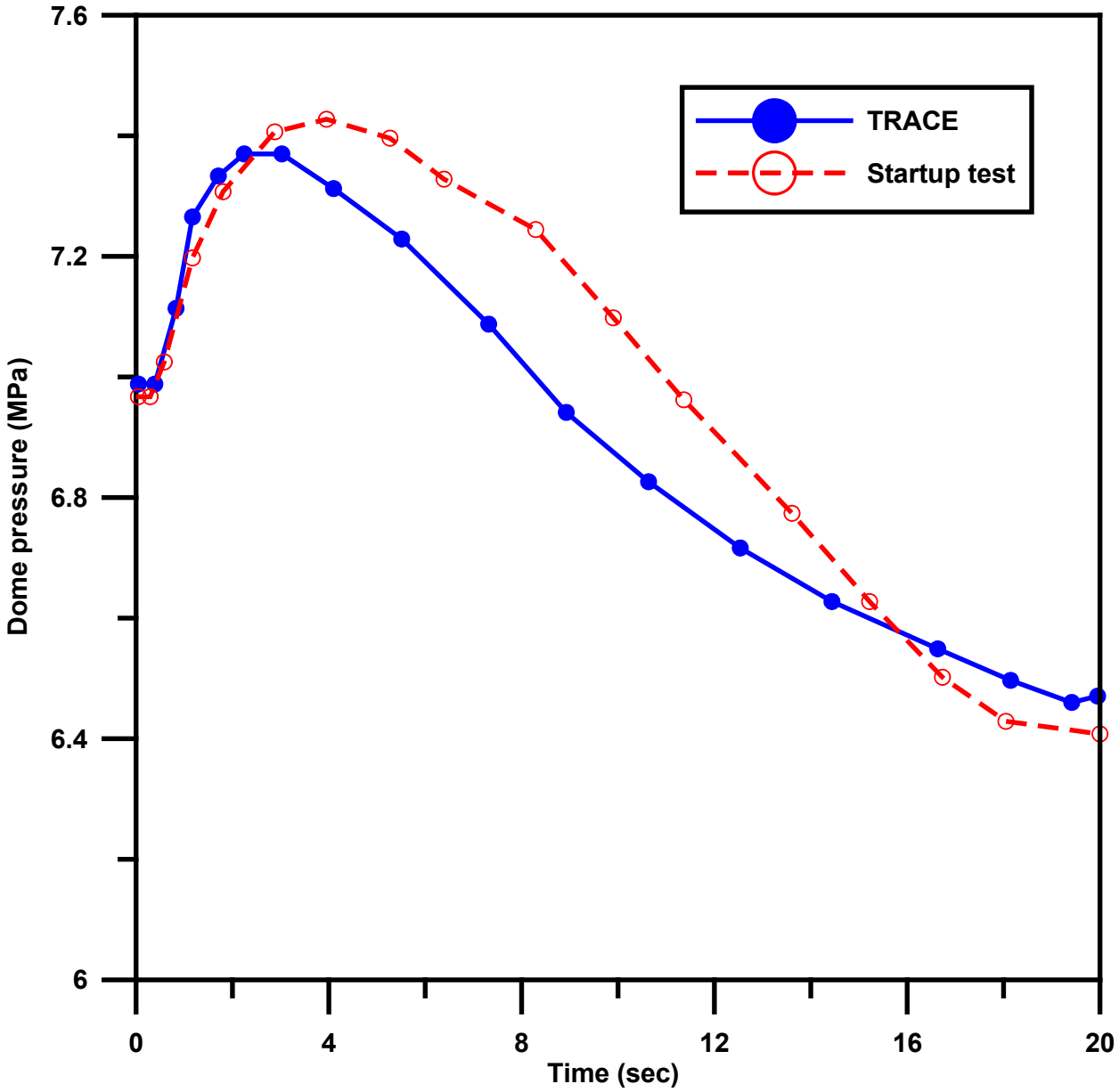


Figure 16 The dome pressure data of TRACE and startup test

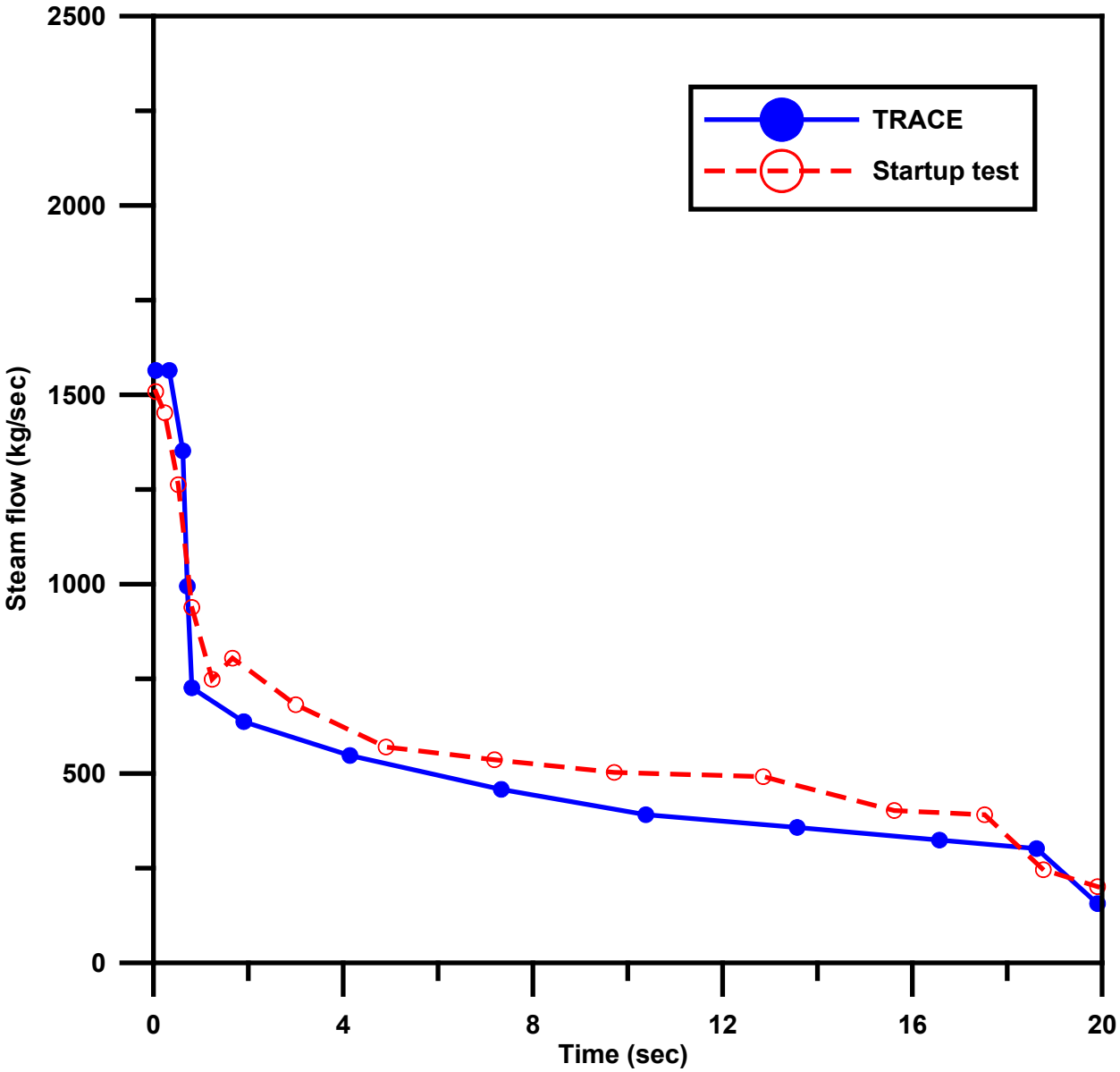


Figure 17 The steam flow data of TRACE and startup test

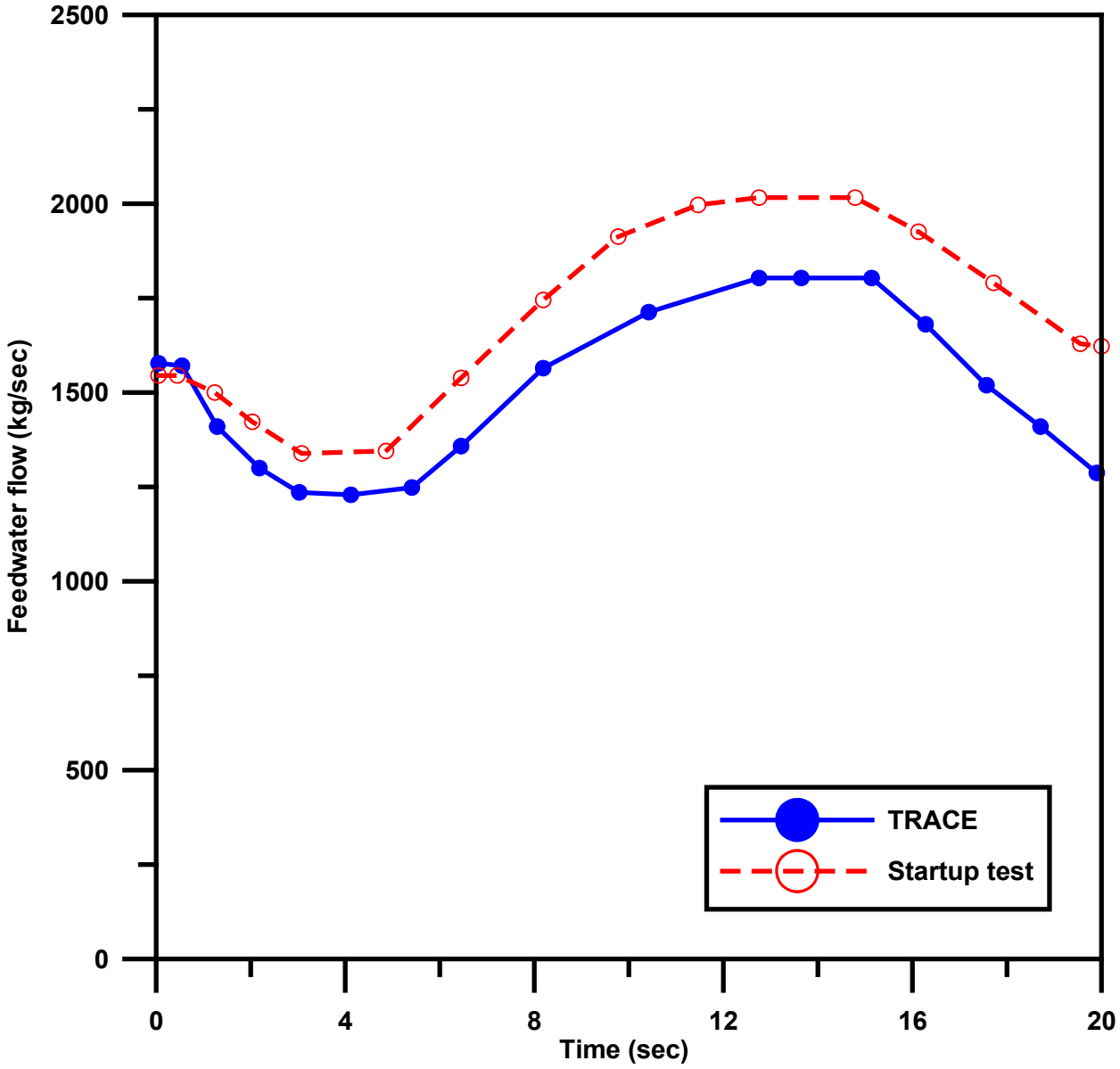


Figure 18 The feedwater flow data of TRACE and startup test

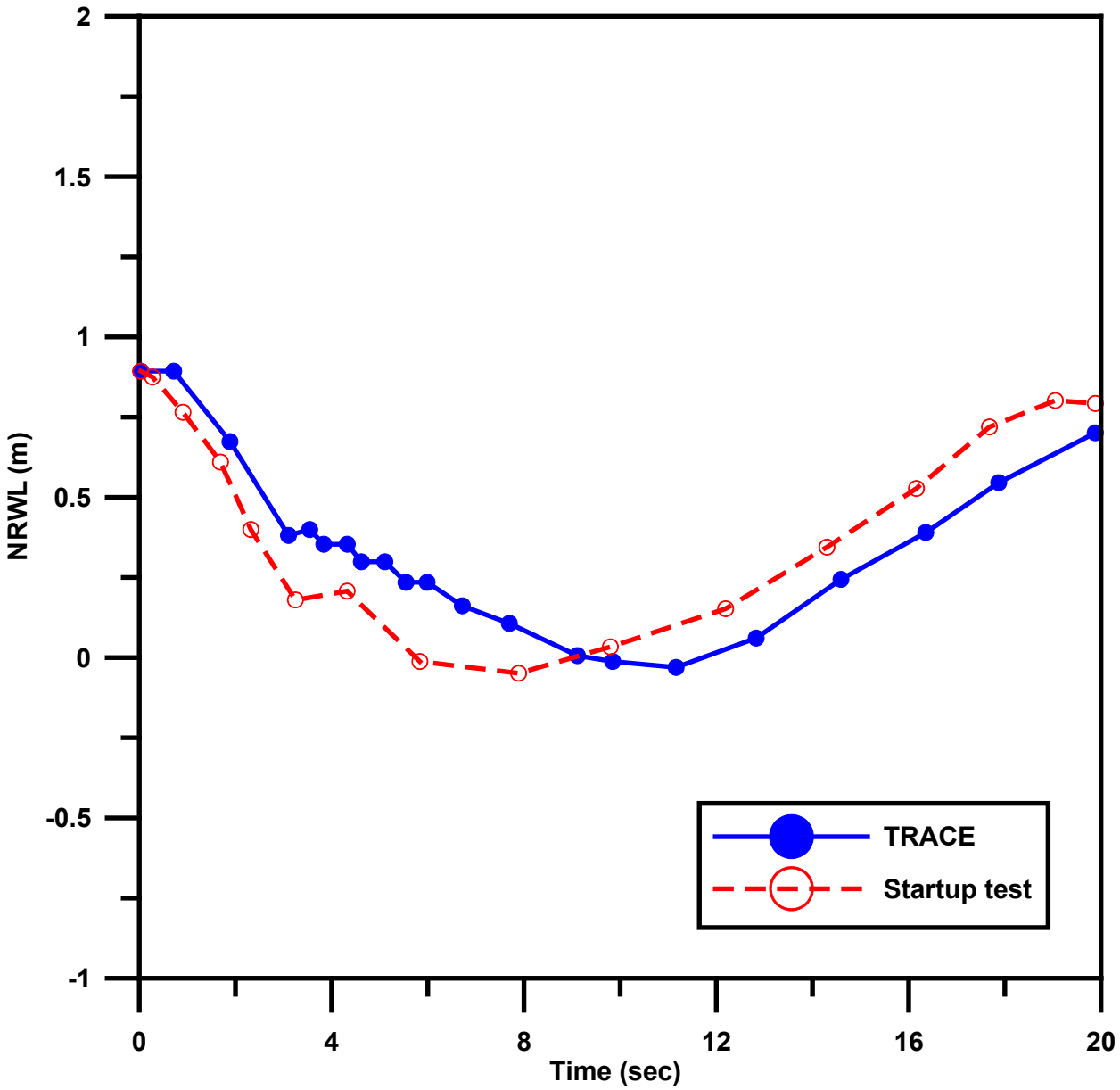


Figure 19 The narrow range water level (NRWL) data of TRACE and startup test

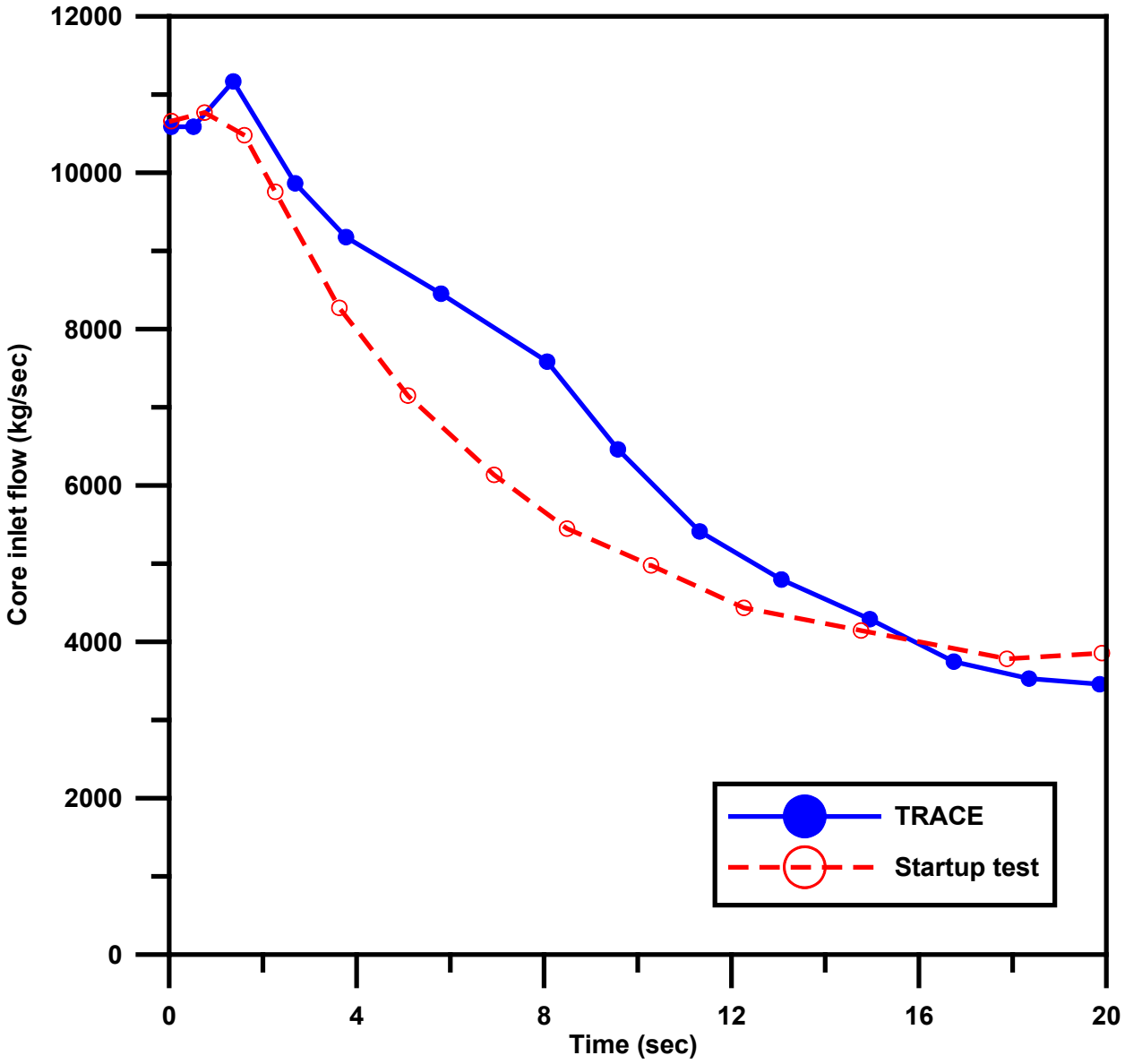


Figure 20 The core inlet flow data of TRACE and startup test

3.2 Startup Test : One Feedwater Pump Trip

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

Fig.21~24 shows the results of startup test and TRACE. Fig. 21 depicts the core inlet flow curves of startup test data and TRACE. The trends of their curves are similar. One feedwater pump trip caused the water level decrease. The FCV runback was tripped when the water level reached level 4. Therefore, the core inlet flow decreased due to the FCV runback. Fig. 22 compares the powers of startup test and TRACE. The trends of the curves are approximately in agreement. After FCV runback, the power decreased. Fig. 23 shows the feedwater flow data of startup test and TRACE. The TRACE result was consistent with the startup test data. One feedwater pump trip caused the feedwater flow to decrease after 4.9 sec. Besides, the NRWL result of TRACE was similar to startup test data (shown in Fig. 24). In summary, the results of TRACE prediction were similar to startup test data 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

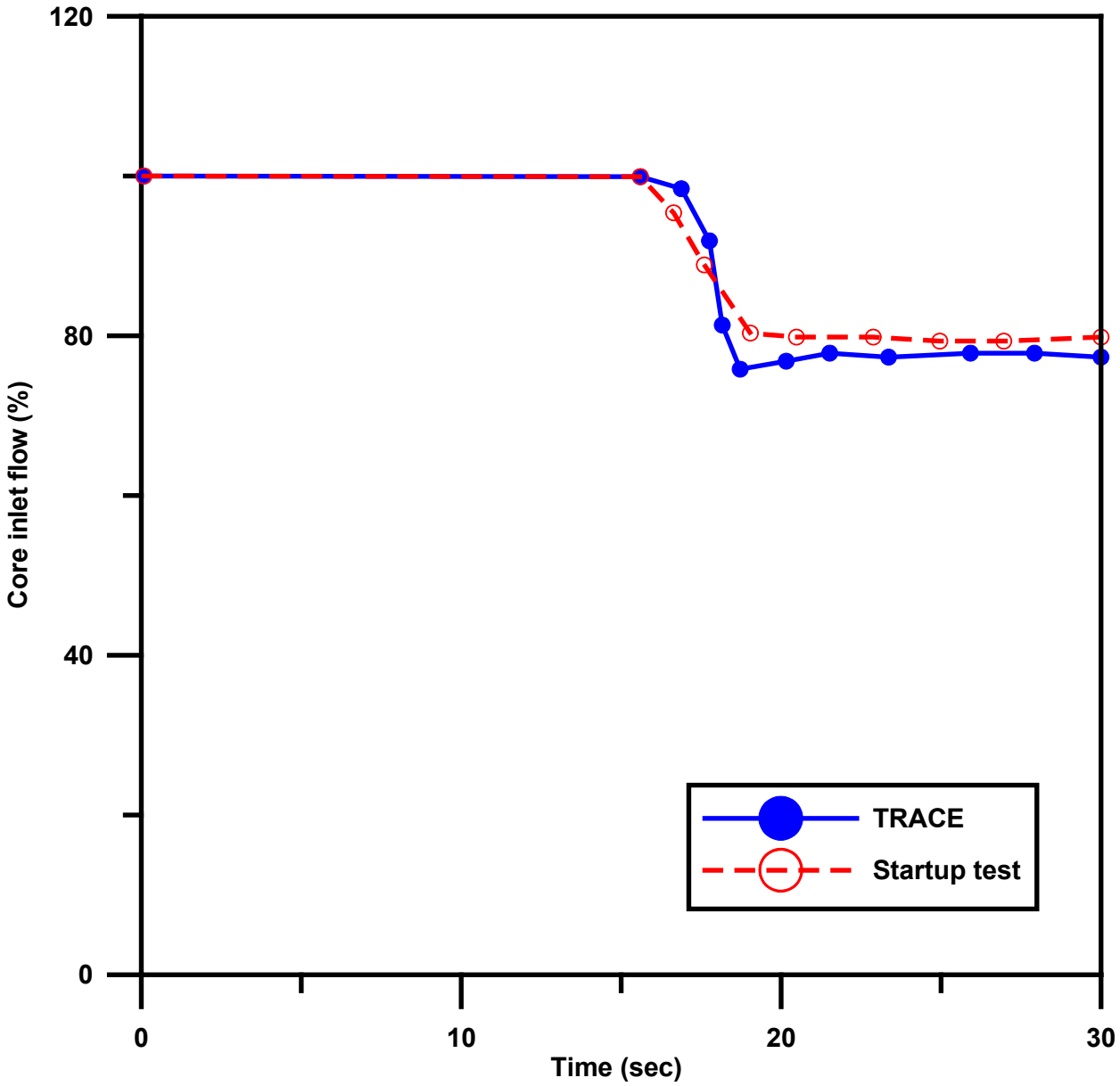


Figure 21 The core inlet flow data of TRACE and startup test

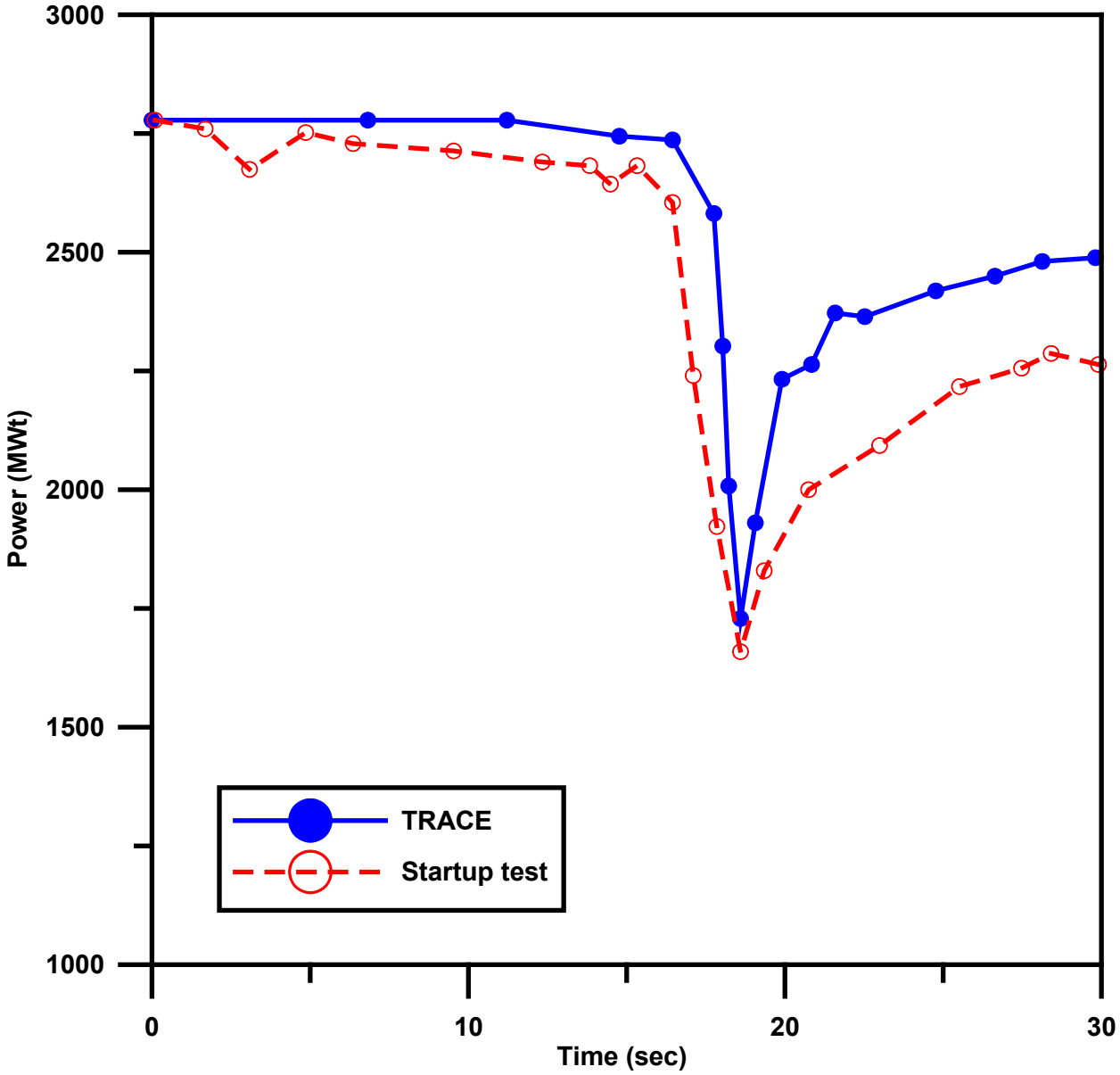


Figure 22 The power data of TRACE and startup test

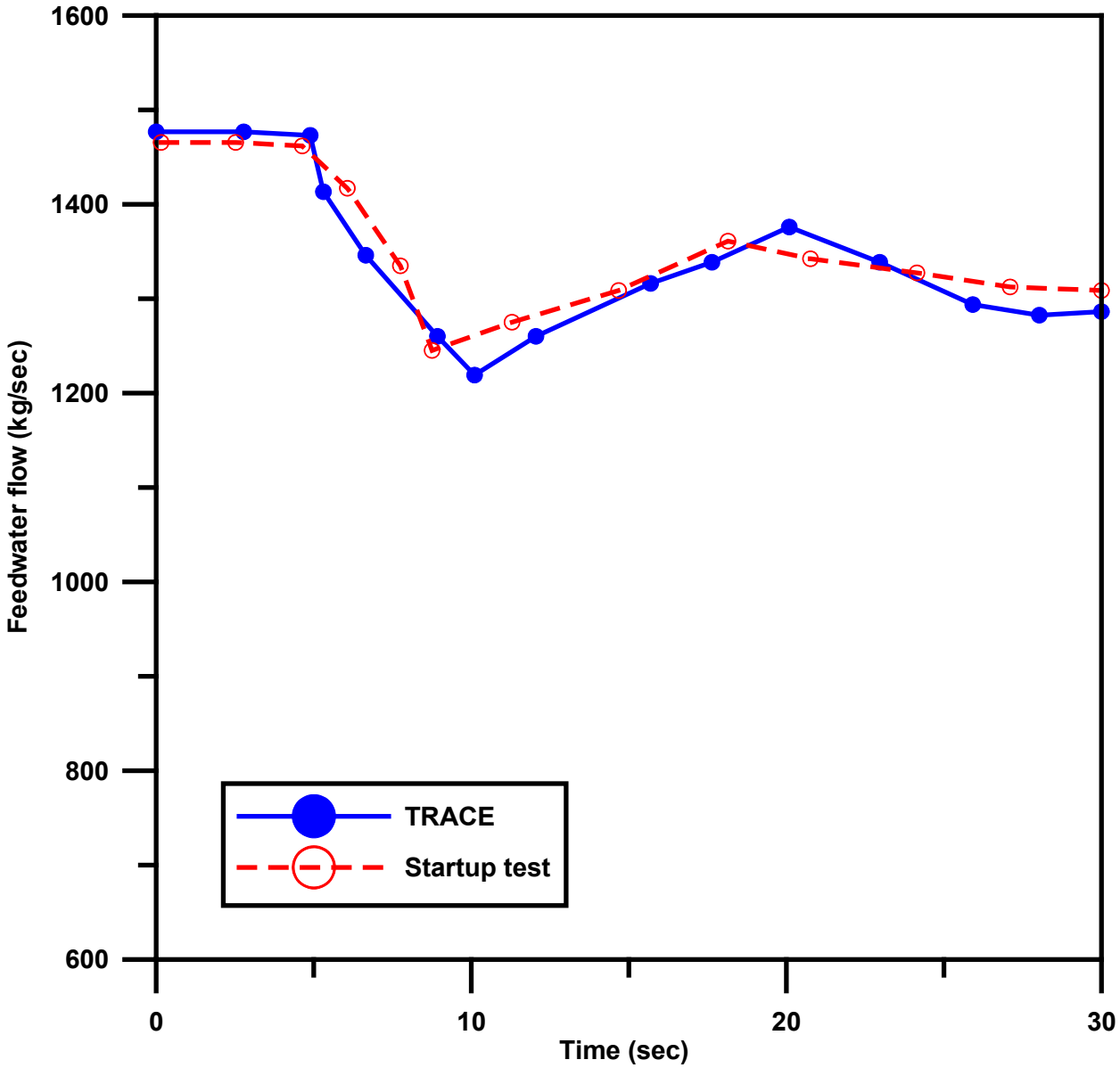


Figure 23 The feedwater flow data of TRACE and startup test

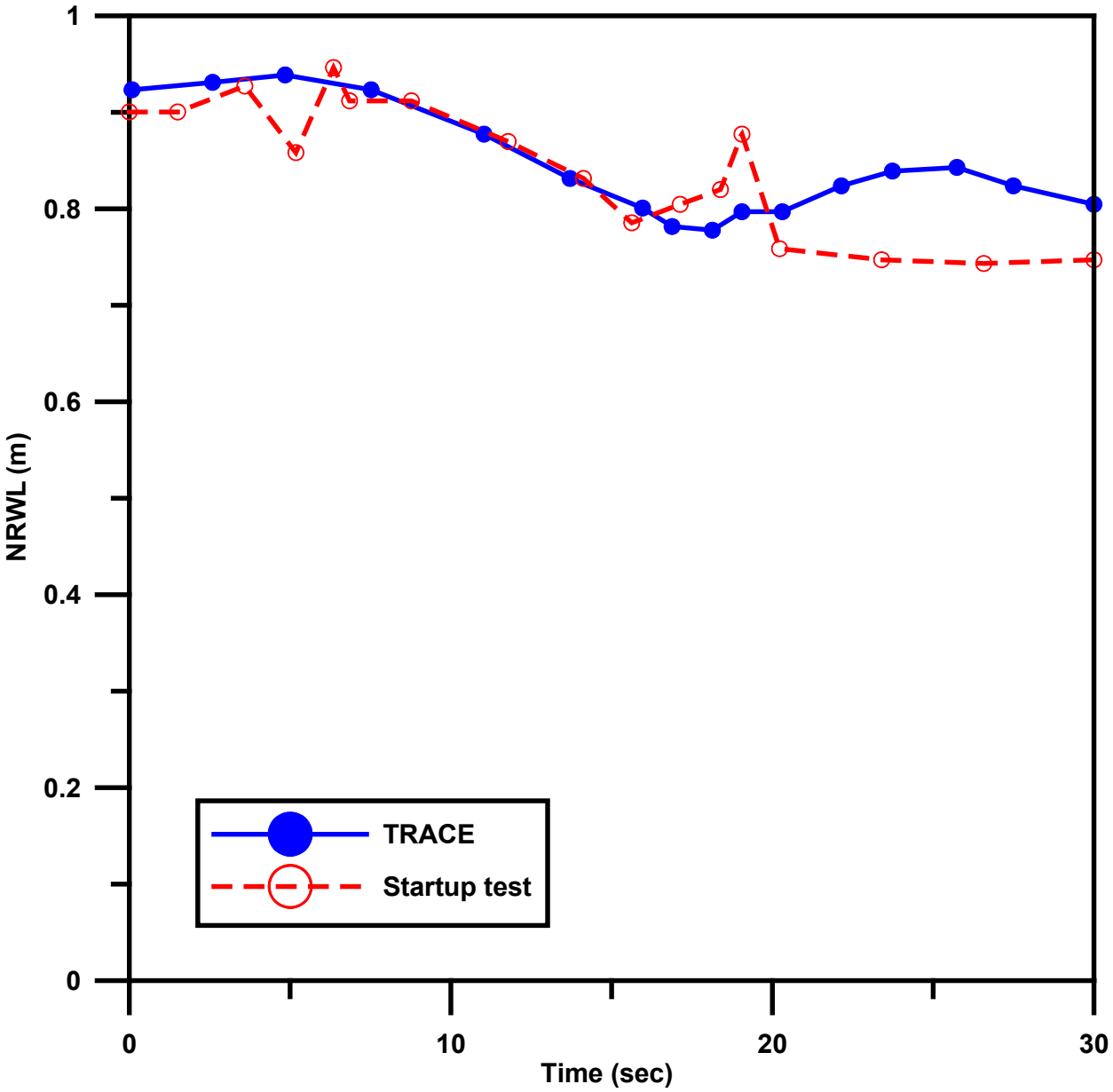


Figure 24 The NRWL data of TRACE and startup test

3.3 FSAR Case : Generator Load Rejection with Bypass Valves

This transient is similar to the startup test: generator load rejection with bypass valves. The initial power was 2894 MWt in this transient. Table 4 compares the generator load rejection with bypass valves transient's sequences of FSAR data with TRACE. Their sequences are very similar. In this transient, when load rejection occurred, the TCV closed quickly. Then the recirculation pump tripped, reactor scrammed, BPV opened, and loss of feedwater generated.

Fig. 25~28 shows the results of FSAR and TRACE. Fig. 25 depicts the power curves of FSAR and TRACE. The trend of TRACE was consistent with FSAR data. The TCV fast closure caused the dome pressure increase which made the core void fraction decrease and power increase. Then, TCV fast closure tripped the reactor scram. Therefore, the power dropped after 0.7 sec. Fig. 26 compares the steam dome pressures of FSAR and TRACE. The trends of their curves were similar. The TCV closing caused the dome pressure to rise. Then, BPV and SRVs opened and led to the decline of dome pressure. Fig. 27 shows the steam flow data of FSAR and TRACE. The TRACE result was consistent with the FSAR data. The TCV closing caused the steam flow to drop sharply. The BPV and SRVs opened subsequently which resulted in the steam flow increase. Due to the dome pressure increase, it resulted in the core inlet flow rising during 0~0.25 sec (see Fig. 28). Then, recirculation pumps trip caused the decrease of core inlet flow. In summary, TRACE prediction results were consistent with FSAR data but there were a few differences in the values of the prediction. Because we cannot find the detailed FSAR data, we don't know what the reasons cause the differences of TRACE results and FSAR data.

Table 4 The comparison of sequences between FSAR and TRACE

Action (sec)	FSAR	TRACE
Transient started	0	0
TCV started to close	0	0
Recirculation pump tripped	0.25	0.25
Reactor scrammed	0.6	0.69
Bypass valve opened	1.7	2.1
Loss of feedwater	3.6	3.5
End of analysis	20	20

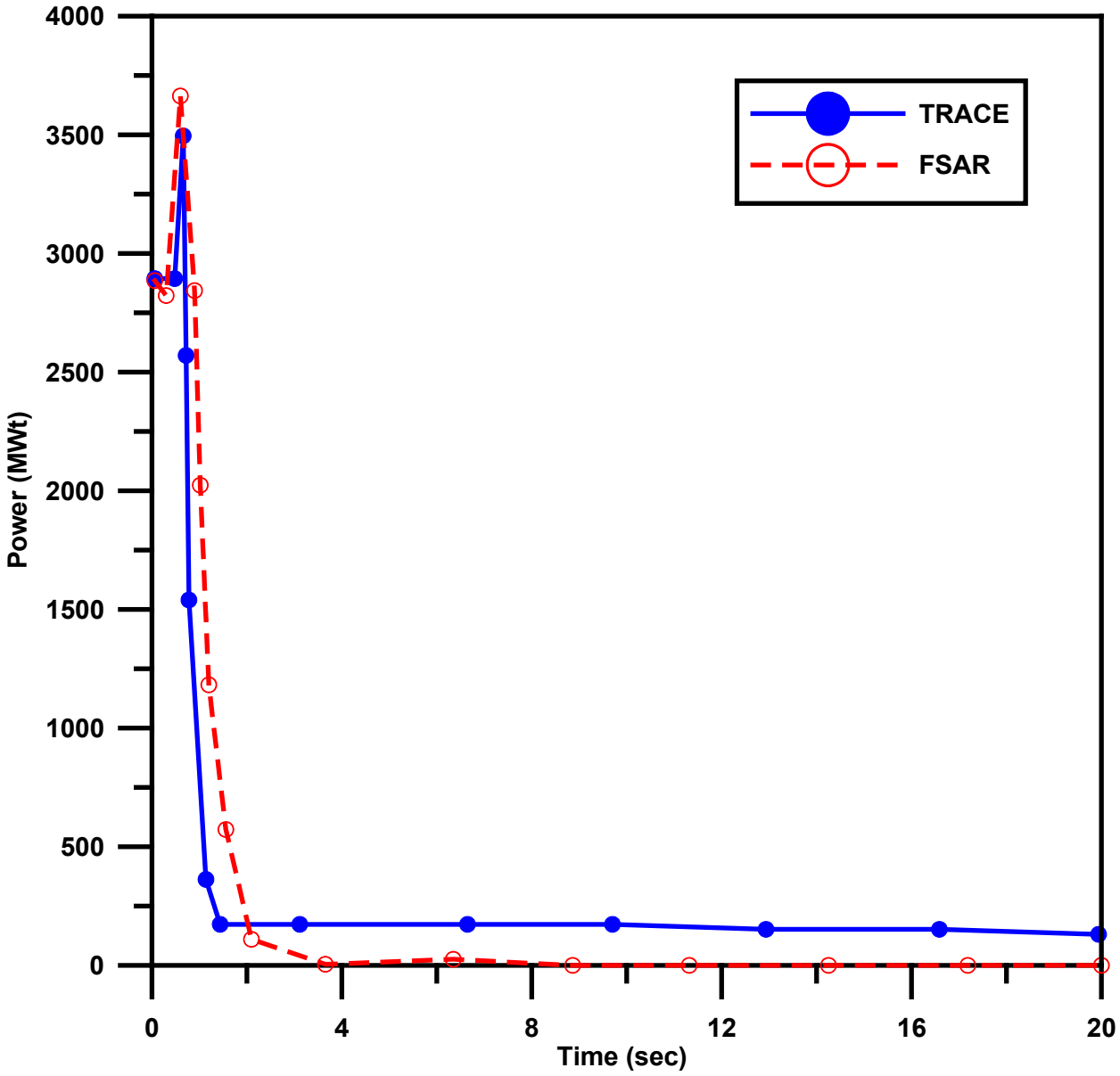


Figure 25 The power data of TRACE and FSAR

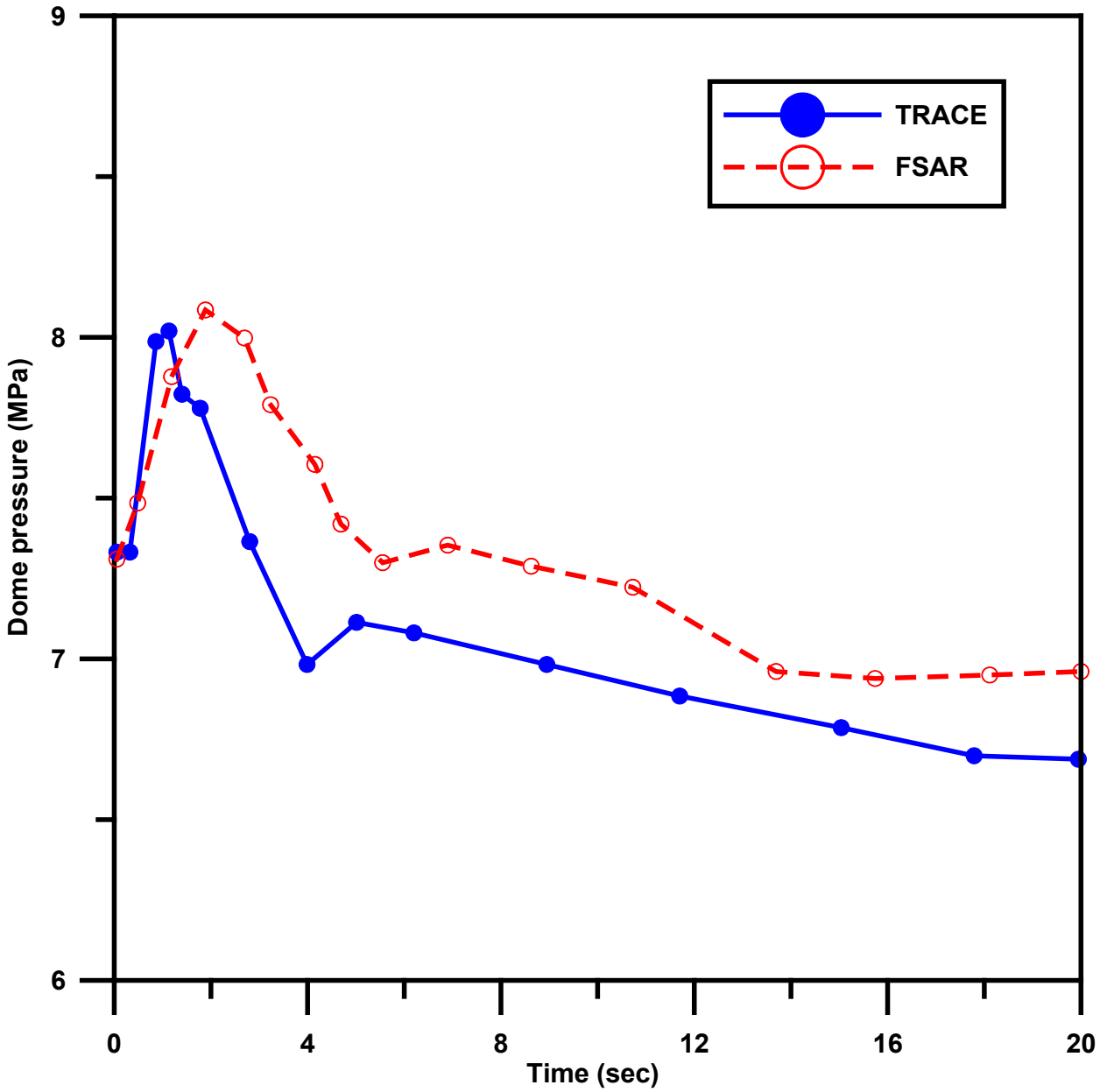


Figure 26 The dome pressure data of TRACE and FSAR

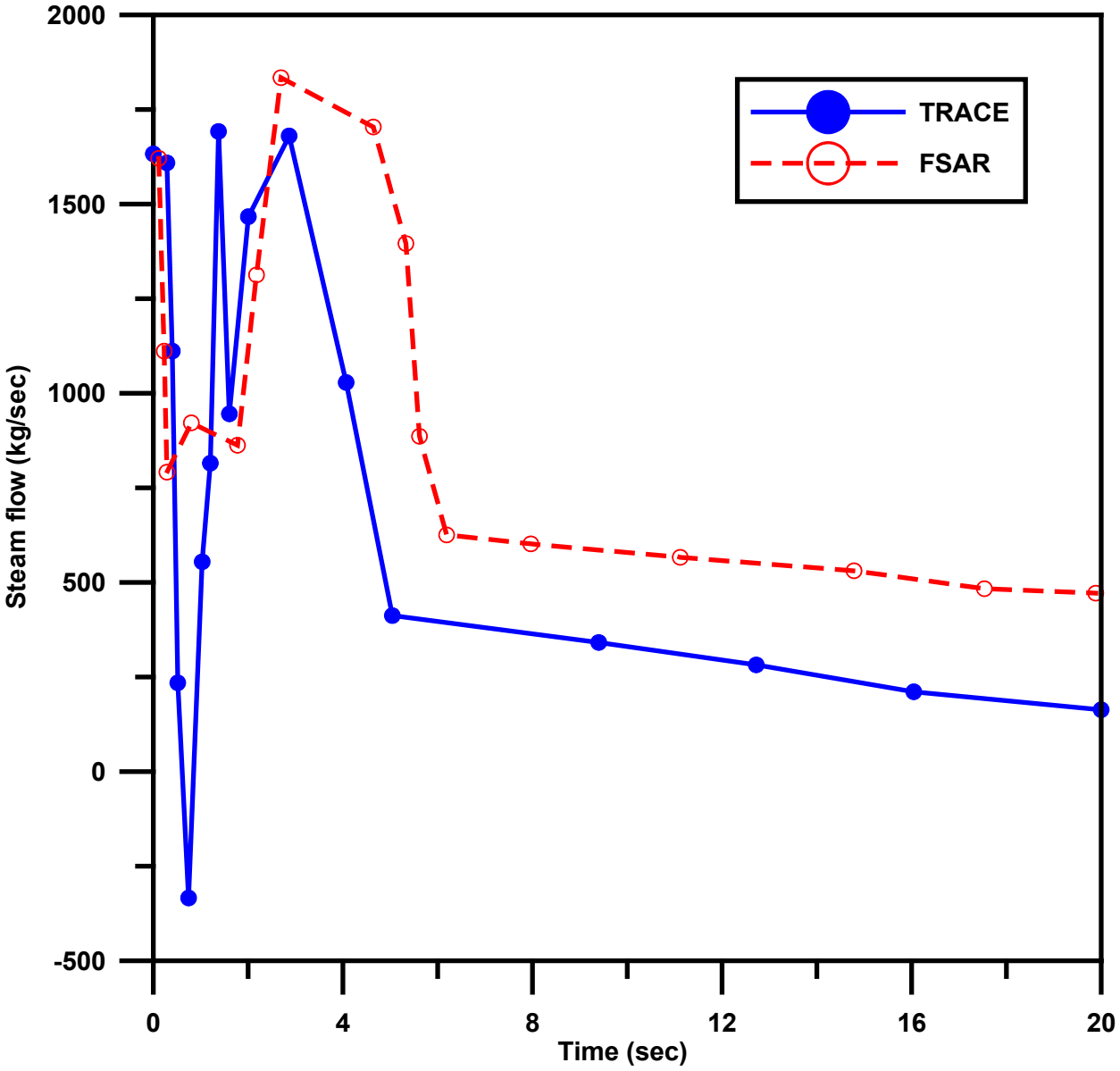


Figure 27 The steam flow data of TRACE and FSAR

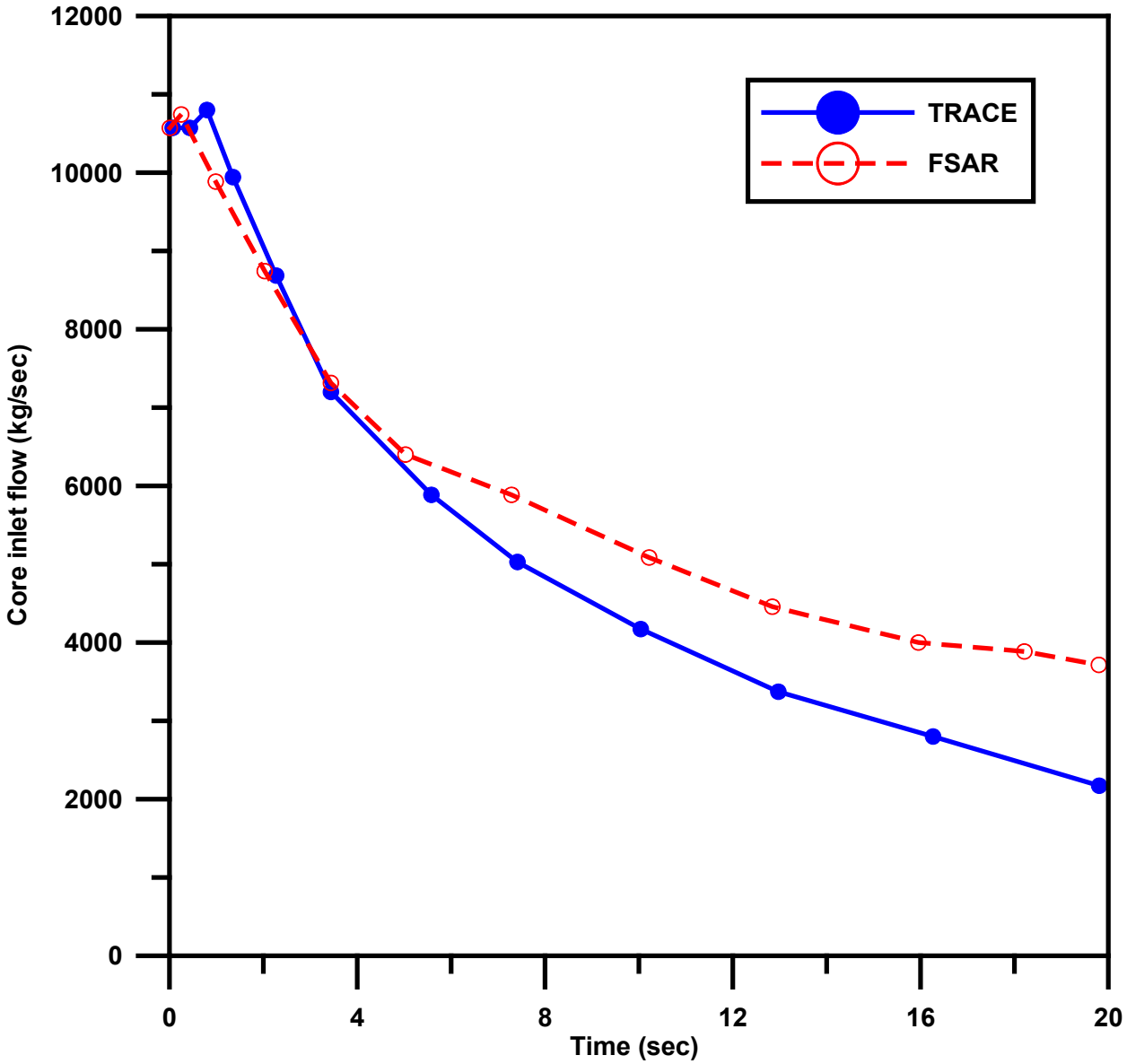


Figure 28 The core inlet flow data of TRACE and FSAR

3.4 Core Shroud Leakage Analysis

There are six assumed cases in the Kuosheng NPP TRACE analysis, as follows:

Case 1: the core shroud leakage occurred (only in 1 azimuthal sector of the vessel, a small hole), the length of break is 0.4 m, the area of break is 0.004 m^2.
Case 2: the core shroud leakage occurred (only in 1 azimuthal sector of the vessel, a small hole), the length of break is 0.4 m, the area of break is 0.014 m^2.
Case 3: the core shroud leakage occurred (in 2 azimuthal sectors of the vessel, 360 degree larger break), the area of break is 10.248 m^2.
Case 4: the case 3 condition + SBO + scram failed + RCIC failed.
Case 5: the case 3 condition + SBO + LOCA (one steamline break) + scram failed + RCIC failed.
Case 6: the case 3 condition + SBO + LOCA (one steamline break).

The initial conditions of cases are 100% rated power/100% rated core flow and the sequences of case 1~6 are shown in Table 5. First, the simulation of steady state is performed during 0~200 sec for all cases. Second, the core shroud leakage started to occur at 200 sec for the cases. Besides, the SBO transients (ex: recirculation pump, feedwater, turbine tripped and MSIV closure) were performed for case 4~6 and the LOCA transients (one steamline break) were performed for case 5~6. Fig. 29~33 show the results of TRACE for all cases.

Fig. 29 show the power results of TRACE. In all cases, when core shroud leakage occurred, the power increased rapidly. Then the power dropped sharply due to the void fraction increase. After core shroud leakage generated fully, the powers of case 1~3 reached the new steady state. However, case 4 and 5 had more negative Doppler reactivity due to the fuel temperature raise (shown in Fig. 32). So the power of case 4 and case 5 was lower than the power case 1~3 after core shroud leakage generated fully. Because the reactor scram occurred, the power of case 6 dropped sharply.

Fig. 30 shows the water level results of Kuosheng NPP core shroud leakage cases. In case 2~6, when core shroud leakage occurred, the water level went up rapidly and was larger than level 8. It also indicated that the level 8 signal was tripped. Therefore, if the Kuosheng NPP finds the level 8 signal tripped, it may be the core shroud leakage happened. Besides, the feedwater tripped and RCIC failed resulted in no water injection to the vessel for case 4 and 5. Due to no water supply, the water level of the case 4 and case 5 decreased continuously and were lower than the TAF (see Fig. 31). Due to one steamline break, the larger steam rushed out for case 5 and 6 that also caused the water level lower than the TAF. The above results caused the fuel temperature increase for the case 4~6 (shown in Fig. 32). The fuel temperature of the case 4 did not reach 1088K [6] during 0~1000 sec (when the temperature is larger than 1088 K, the zirconium-water reaction may generate). However, if the time of the transient is enough long, we think that the fuel temperature of the case 4 may be larger than 1088 K. Besides, because the fuel temperature of the case 5 was larger than 1088 K, the zirconium-water reaction happened. The above results indicated that the fuels might be damaged after the zirconium-water reaction happened. In case 6, the SBO + LOCA also occurred which was like case 5. However, the reactor scram and RCIC were active in case 6. Therefore, the fuel temperature increase was delay. The fuel temperature of case 6 went up after 500 sec. It indicated that only RCIC injection was not enough for case 6. Fig. 33 shows the dome pressures of TRACE for all cases. There were not variation in case 1~3. In case 4, because SRVs opened, there was an oscillation in this transient. In case 5~6, because the LOCA occurred, the dome pressure dropped sharply. Finally, Fig. 34 depicts the animation model of Kuosheng NPP for case 5.

Table 5 The sequences of case 1~6 for core shroud leakage transient

Action (sec)	Case 1	Case 2	Case 3
Transient started	200	200	200
Core shroud leakage started	200 (leakage area: 0.004m²)	200 (leakage area: 0.14m²)	200 (leakage area: 10.248m²)
NRWL reached level 8	-	208	204
End	1000	1000	1000

Action (sec)	Case 4 (SBO)	Case 5 (SBO+LOCA)	Case 6 (SBO+LOCA)
Transient started	200	200	200
Core shroud leakage started	200 (leakage area: 10.248m²)	200 (leakage area: 10.248m²)	200 (leakage area: 10.248m²)
Steam line LOCA generated	-	200	200
TCV started to close	200.1	200.1	200.1
Feedwater tripped	200.1	200.1	200.1
Reactor scrammed	-	-	200.1
Recirculation pmup tripped	200.1	200.1	200.1
MSIV started to close	200.1	200.1	200.1
Water level reached level 8	204	204	204
RCIC started	-	-	234
Water level reached TAF	261	260	291
Water level reached BAF	-	431	-
Cladding temperature reached 1088K	-	550	-
End	1000	1000	1000

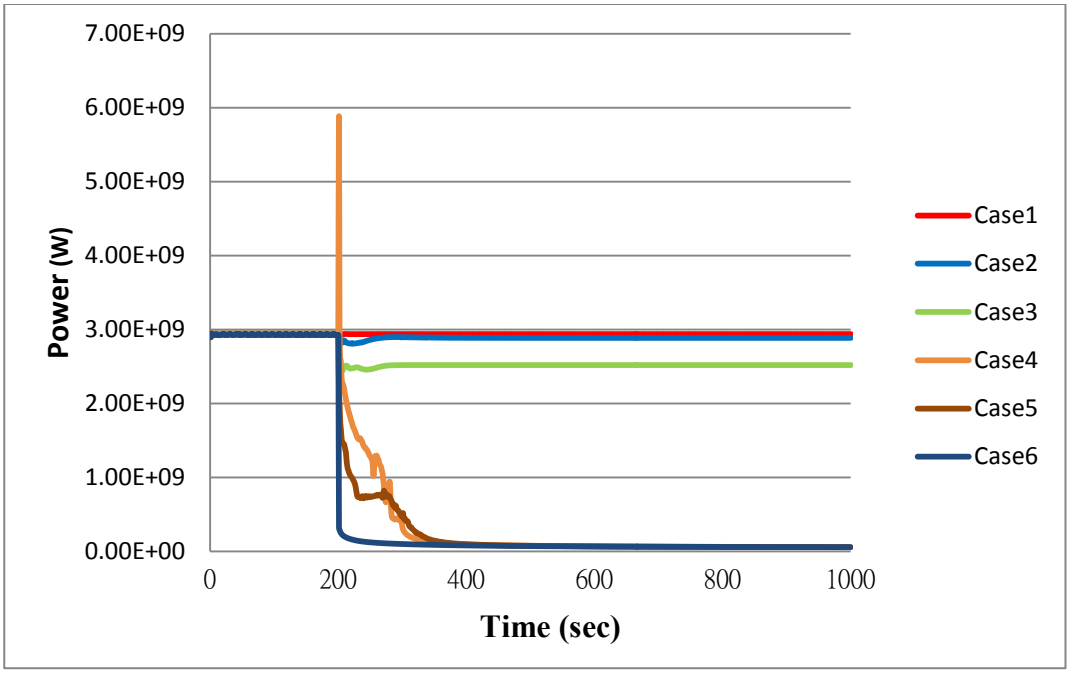


Figure 29 The power results

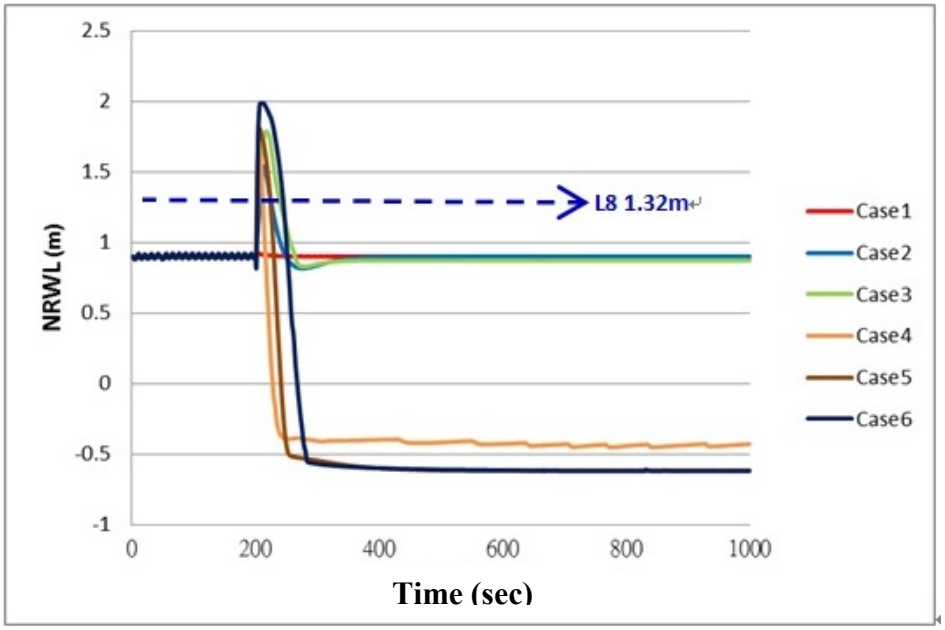


Figure 30 The NRWL results

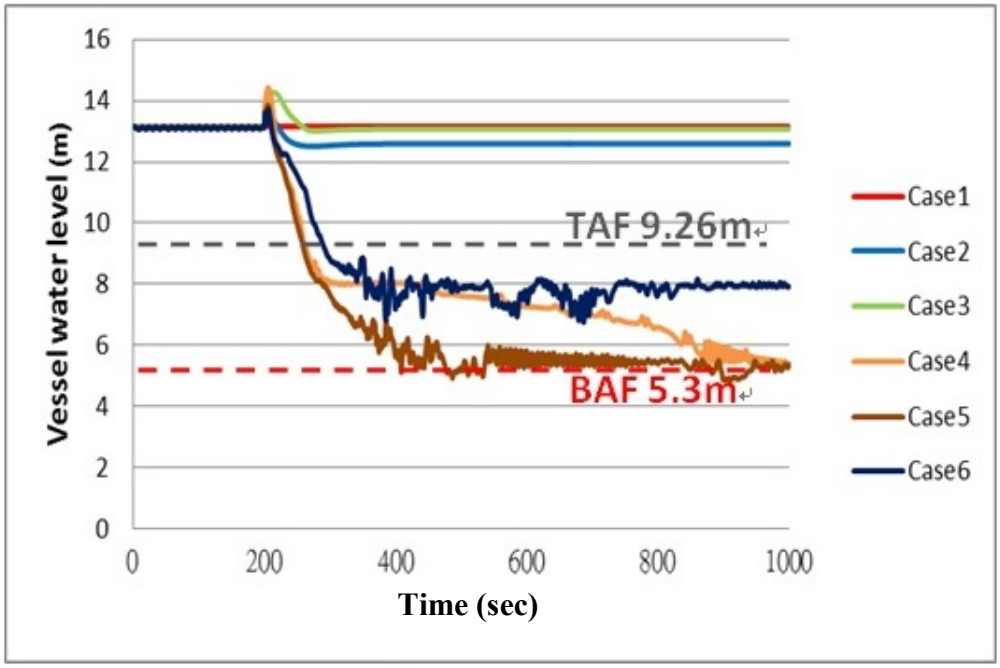


Figure 31 The vessel water level results

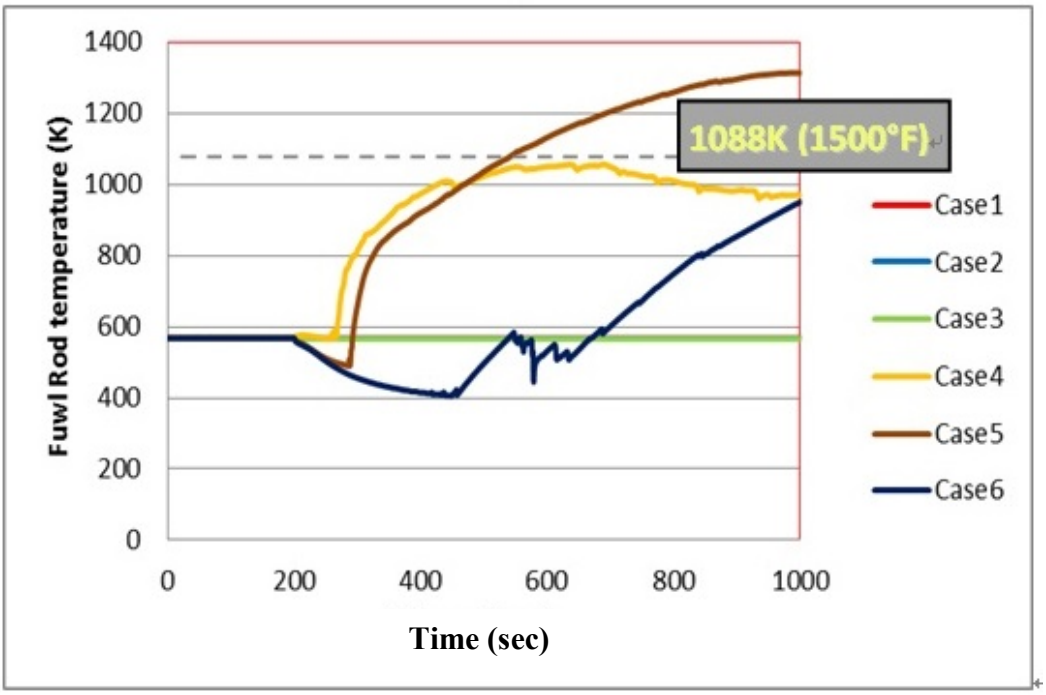


Figure 32 The fuel rod temperature results

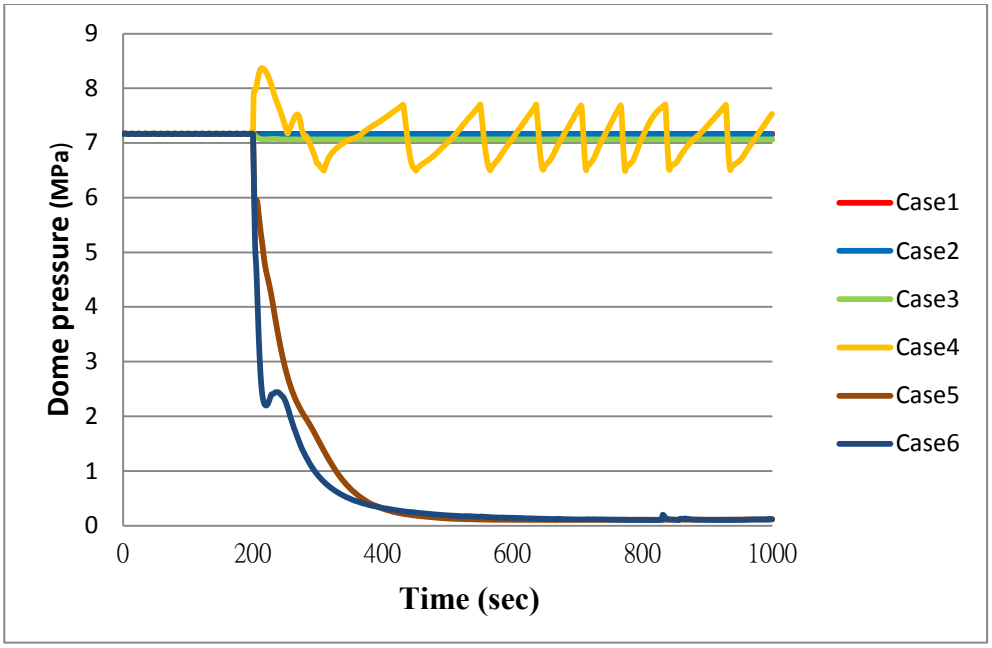
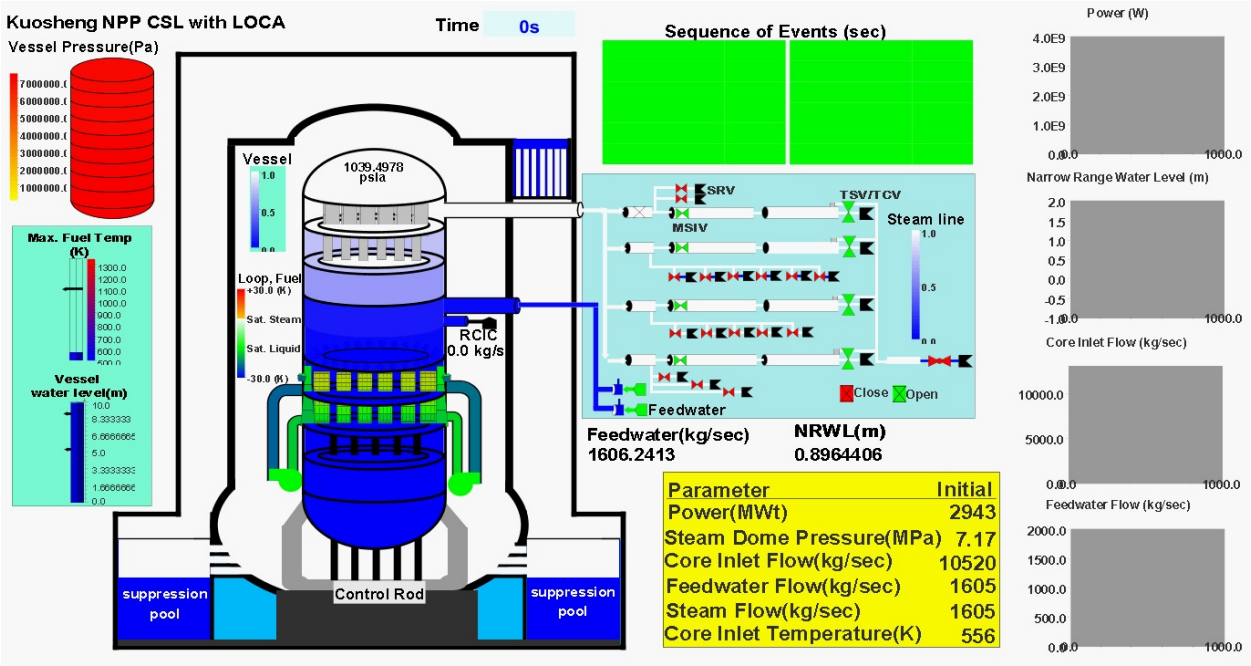
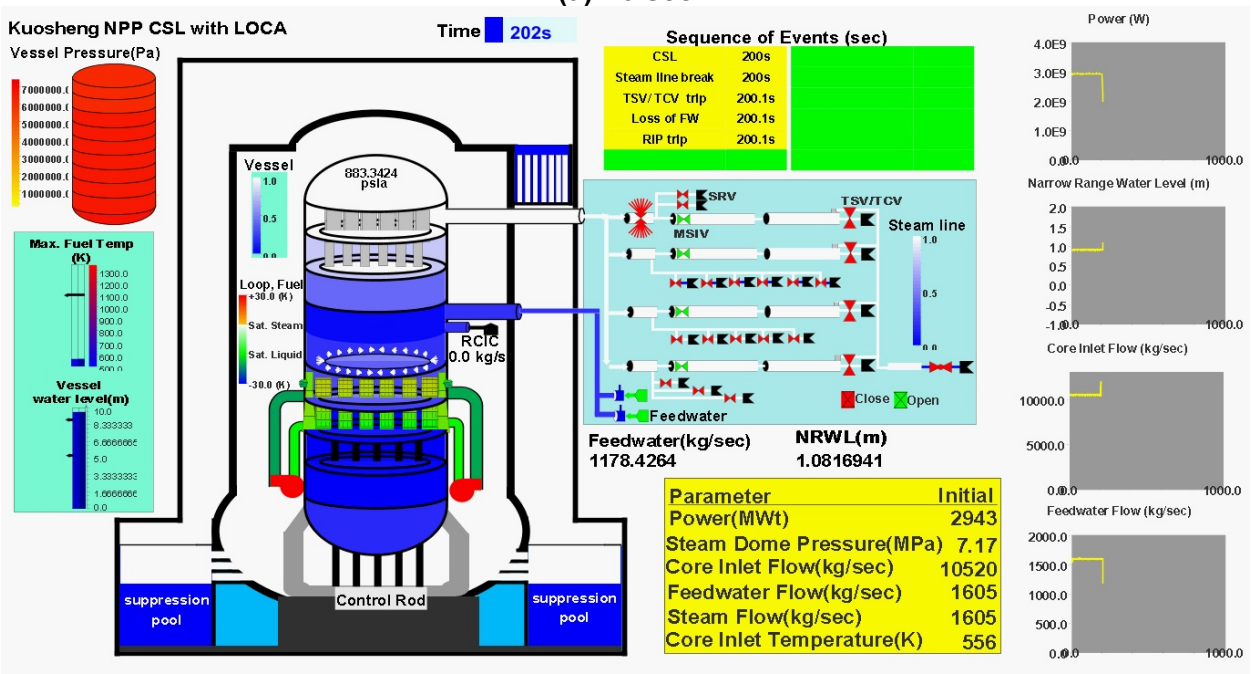


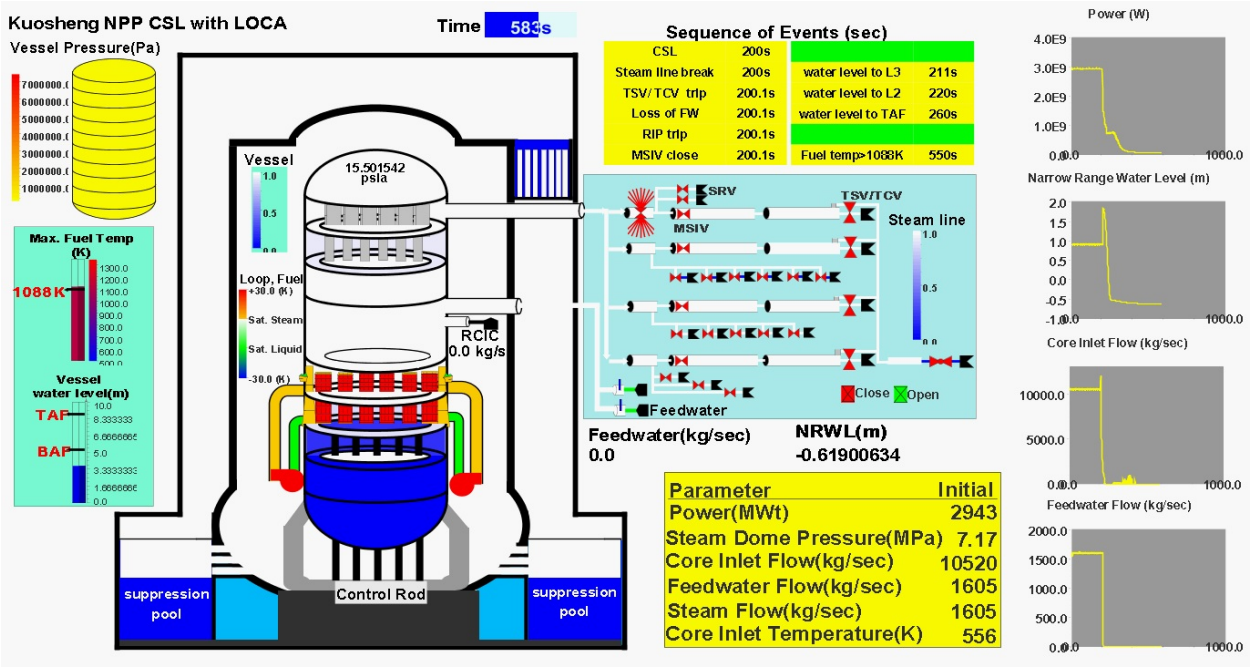
Figure 33 The dome pressure results



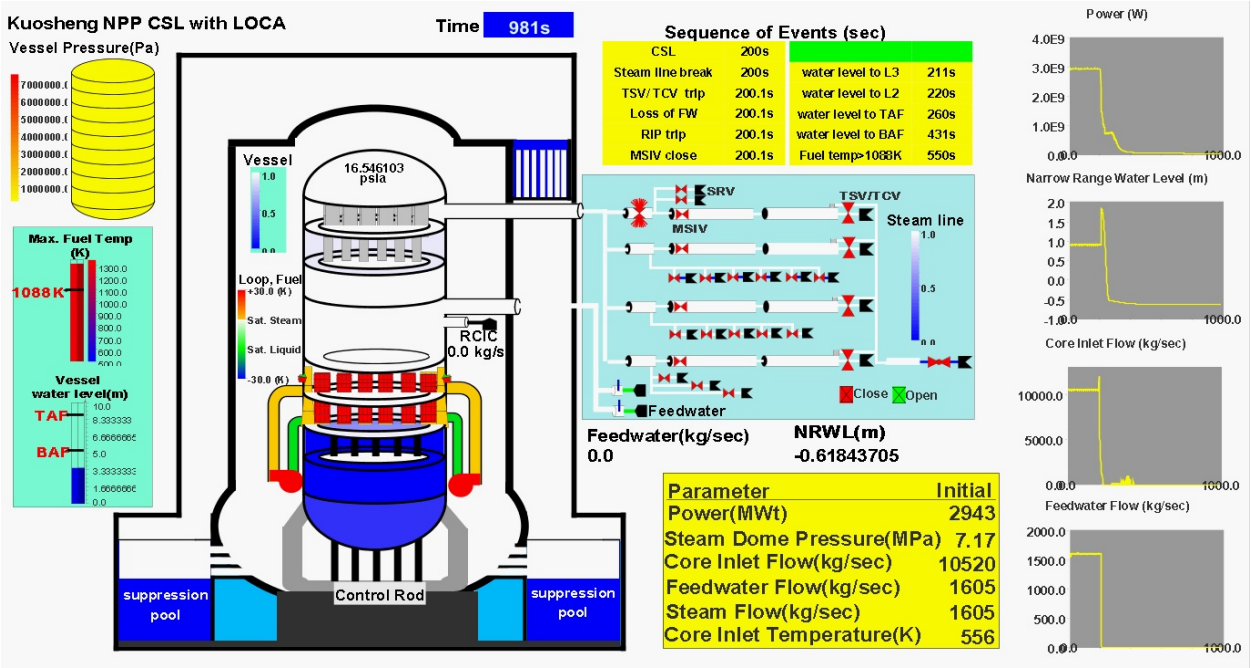
(a) 0 sec



(b) 202 sec



(c) 583 sec



(d) 981 sec

Figure 34 The animation model of Kuosheng NPP for core shroud leakage transient (a) 0 sec, (b) 202 sec, (c) 583 sec, (d) 981 sec

4. CONCLUSIONS

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

This study also developed the TRACE/SNAP core shroud leakage models of Kuosheng NPP. There are six assumed cases in this research and the results of TRACE show the accurate response of system for these cases. Besides, the results of TRACE indicated that the Kuosheng NPP may find the core shroud leakage happened by the level 8 signal tripped. The TRACE analysis results also show that the pure core shroud leakage transient (case 1~3) wasn't influence the Kuosheng NPP safety. However, the core shroud leakage + station blackout (SBO) / one steamline break (LOCA) transient (case 4~6) may cause the cladding temperature larger than 1088 K and influence the Kuosheng NPP safety.

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10. SUPPLEMENTARY NOTES
K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

Kuosheng Nuclear Power Plant (NPP) is the second NPP in Taiwan which is the BWR/6 plant. This research focuses on the development of the Kuosheng NPP TRACE/SNAP model. In order to check the system response of the Kuosheng NPP TRACE/SNAP model, this study uses the analysis results of Final Safety Analysis Report (FSAR) and startup tests data to assess the Kuosheng NPP TRACE/SNAP model. The load rejection and a feedwater pump trip transients were selected to validate the Kuosheng NPP TRACE/SNAP model. The trends of TRACE analysis results were consistent with the FSAR and startup tests data. It indicates that there is a respectable accuracy in the Kuosheng NPP TRACE/SNAP model. Besides, this research also focuses on the application of the Kuosheng NPP TRACE/SNAP model in the core shroud leakage. The core shroud leakage is one of issues of concern by the U.S. NRC and NPPs. This research is using the Kuosheng NPP TRACE/SNAP model to perform the core shroud leakage transients for Kuosheng NPP safety analysis. The TRACE analysis results show that the pure core shroud leakage transient wasn't influence the Kuosheng NPP safety. However, the core shroud leakage + station blackout (SBO) / steamline break (loss of coolant accident, LOCA) transient caused the cladding temperature larger than 1088 K and affected the Kuosheng NPP safety.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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TRACE/SNAP
Measurement Uncertainty Recovery (MUR)
Measurement Uncertainty Recovery Power Uprate (MURPU)
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