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The Establishment and Assessment of Chinshan (BWR/4) Nuclear Power Plant TRACE/SNAP Model

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ABSTRACT

Chinshan Nuclear Power Plant (NPP) is the first NPP in Taiwan which is of the BWR/4 type plant. This research focuses on the development of the Chinshan NPP TRACE/SNAP model. In order to check the system response of the Chinshan NPP TRACE/SNAP model, this study uses the analysis results of Final Safety Analysis Report (FSAR) to assess the Chinshan NPP TRACE/SNAP model. The increase in reactor pressure transients including turbine trip and main steam isolation valves closure were selected to validate the Chinshan NPP TRACE/SNAP model. The trends of TRACE analysis results were consistent with the FSAR data. It indicates that there is a credible fidelity in the Chinshan NPP TRACE/SNAP model. In addition, this research also investigates the application of the Chinshan NPP TRACE/SNAP model for the core shroud leakage. The core shroud leakage is one of current issues of concern of the U.S. NRC and other BWR/4 NPP owners. This research utilizes the Chinshan NPP TRACE/SNAP model to perform the core shroud leakage transients for Chinshan NPP safety analysis. The TRACE analysis results show that the simple core shroud leakage transient does not influence the Chinshan NPP fuel temperature to cause any core damage. However, the core shroud leakage combined with station blackout (SBO) or steamline break (loss of coolant accident, LOCA) transient would cause the cladding temperature to reach higher than 1088 K and affected the Chinshan 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 legacy 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 Chinshan NPP has been built. In this report, this study uses the analysis results of FSAR to assess the Chinshan NPP TRACE/SNAP model.

CONTENTS

FIGURES

TABLES

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 Chinshan NPP is developed by INER.

According to the user manual, TRACE is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool. NRC has ensured that TRACE will be the main code used in thermal hydraulic safety analysis in the future without further development of other thermal hydraulic codes, such as RELAP5 and TRAC. Besides, the 3-D geometry model of reactor vessel, which is one of the representative features of TRACE, can support a more accurate and detailed safety analysis of NPPs. On the whole TRACE provides greater simulation capability than the previous codes, especially for events like LOCA.

Chinshan NPP was building in 1970. It is the first NPP in Taiwan which is the BWR/4 plant and the original rated power for each unit is 1775 MWt. After the project of MUR (Measurement Uncertainty Recovery) for Chinshan NPP, Unit 2 started MURPU (Measurement Uncertainty Recovery Power Uprate) from April 6, 2008 for Cycle 23 and Unit 1 started MURPU from November 8, 2008 for Cycle 24. The operating power is 101.7% of the original designed rated power, which is 1805 MWt now. This research focuses on the development of the Chinshan NPP TRACE/SNAP model. In order to check the system response of the Chinshan NPP TRACE/SNAP model, this study uses the analysis results of FSAR to assess the Chinshan NPP TRACE/SNAP model. The increase in reactor pressure transients including turbine trip and main steam isolation valves closure were selected to validate the Chinshan NPP TRACE/SNAP model. The trends of TRACE analysis results were consistent with the FSAR data. It indicates that there is a respectable accuracy in the Chinshan NPP TRACE/SNAP model. Besides, this research also focuses on the application of the Chinshan 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 Chinshan NPP TRACE/SNAP model to perform the core shroud leakage transients for Chinshan NPP safety analysis. The TRACE analysis results show that the pure core shroud leakage transient wasn't influence the Chinshan NPP safety. However, the core shroud leakage + SBO or steamline LOCA transient caused the cladding temperature larger than 1088 K and affected the Chinshan 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
MUR	Measurement Uncertainty Recovery
MURPU	Measurement Uncertainty Recovery Power Uprate
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
RPV	Reactor Pressure Vessel
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

Chinshan NPP was building in 1970. It is the first NPP in Taiwan which is the BWR/4 plant and the original rated power for each unit is 1775 MWt. After the project of MUR for Chinshan NPP, Unit 2 started MURPU from April 6, 2008 for Cycle 23 and Unit 1 started MURPU from November 8, 2008 for Cycle 24. The operating power is 101.7% of the original designed rated power, which is 1805 MWt now. 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 Chinshan NPP TRACE/SNAP model. In our previous study [2], the preliminary TRACE/SNAP model of Chinshan NPP was established and assessed by the startup tests data. The Chinshan NPP TRACE/SNAP model includes one 3-D vessel, six channels which are used to simulate 408 fuel bundles, four steamlines, and 10 SRVs components, etc.. In order to check the system response of the Chinshan NPP TRACE/SNAP model, this study uses the analysis results of FSAR to assess the Chinshan NPP TRACE/SNAP model. The increase in reactor pressure transients including turbine trip and main steam isolation valves closure were selected to validate the Chinshan NPP TRACE/SNAP model.

Besides, this research also focuses on the application of the Chinshan 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 Chinshan NPP was performed by the Chinshan NPP TRACE/SNAP model in this study. Besides, refer to the conditions of Fukushima NPP event, the core shroud leakage + SBO or LOCA transients were also analyzed by TRACE for Chinshan NPP.

2. MODEL OF CHINSHAN (BWR4) NPP

The code versions adopted in this research are SNAP v 2.2.0 and TRACE v 5.0p3. The process of Chinshan NPP TRACE/SNAP model development is as follows (shown in Fig. 1): First, the system and operating data for the cases of FSAR of Chinshan NPP are collected [1]-[7]. 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 Chinshan NPP. Finally, the Chinshan TRACE/SNAP model is verified with the cases of FSAR. The TRACE/SNAP model of Chinshan NPP is presented in Fig. 2. SNAP also can use the TRACE results data to make an animation for transient, such as Fig. 3.

The reactor vessel is divided into 9 axial levels, 4 radial rings, and 2 azimuthal sectors (shown in Fig. 4). Six channels are used to modeling 408 fuel rods and connect to the vessel's level 3~4. There were 27 axial nodes and 6 radial nodes in one channel. The water rods and partial length rods were also simulated in the channels (shown in Fig. 5). Besides, six separators connect to the level 6~8 of the vessel (see Fig. 6). Two recirculation loops are set outside the vessel, with a recirculation pump in each loop (see Fig. 7). In the TRACE model, 10 groups of injection pumps are merged into an equal injection pump (see Fig. 8). We use valve components to simulate the SRV (safety relief valve), MSIV (main steamline isolation valve), TSV (turbine stop valve), TCV (turbine control valve) and BPV (bypass valve). The critical flow models for the MSIV, SRV, TCV, TSV, and BPV have been considered in our analysis (see Fig. 9). The reactor vessel connects with four steamlines. Besides, every steamline has one MSIV, one TCV/TSV, and 1~4 SRVs. The steam goes through the top of the reactor and into the main steamlines. Finally, the steam passes through the TCVs and enters the turbines (boundary conditions). We also build bypass pipelines and the turbine bypass valve. Moreover, the break component at the end of bypass valve is used to simulate the condenser.

In Chinshan 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. These data are set in the power component (shown in Fig. 10).

In the Chinshan NPP TRACE/SNAP analysis model, there are three simulation control systems included (1) feedwater flow control system, (2) 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 components of TRACE/SNAP.

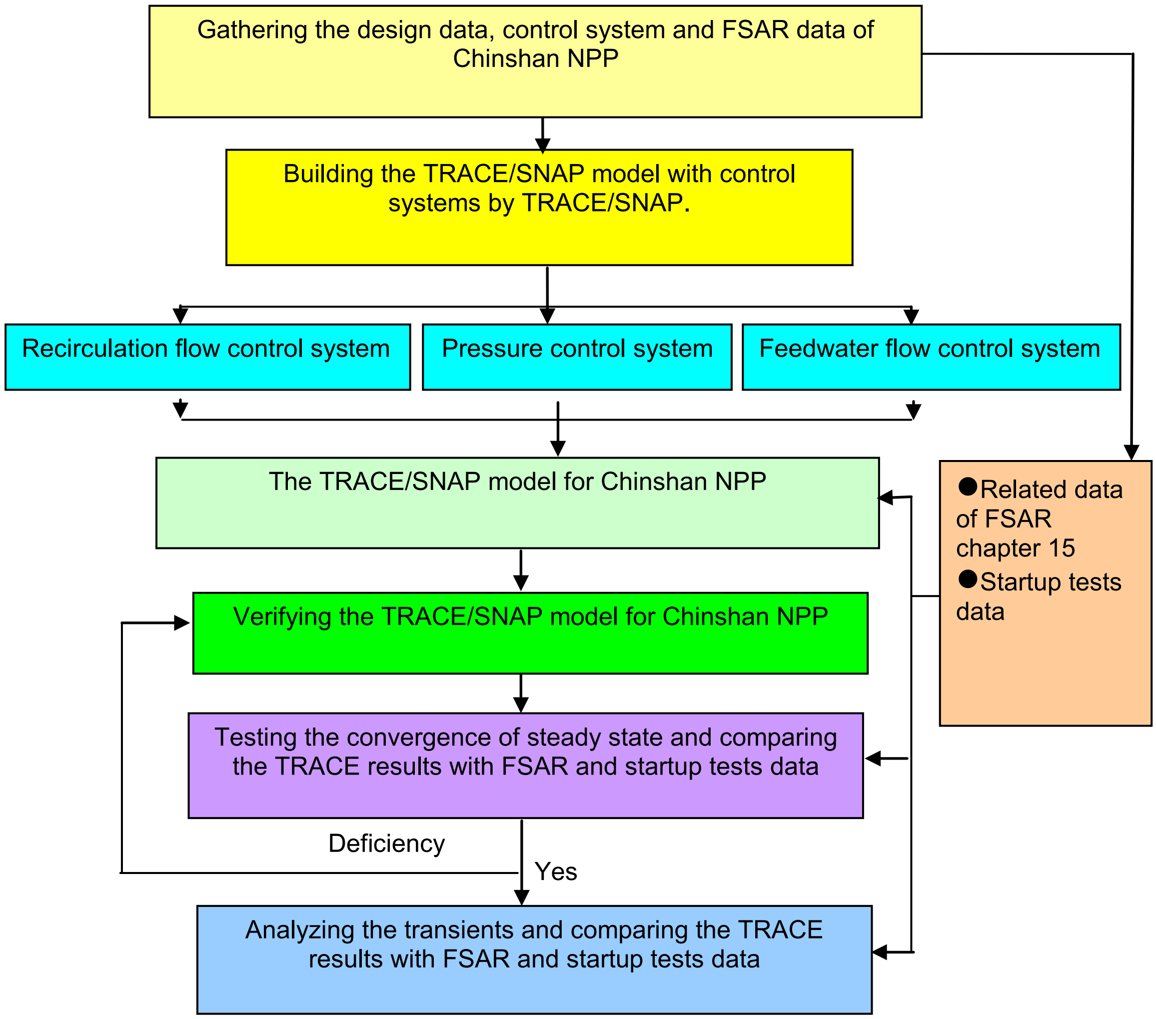

Figure 1 The flow chart of establishing and verifying the TRACE/SNAP model of Chinshan NPP

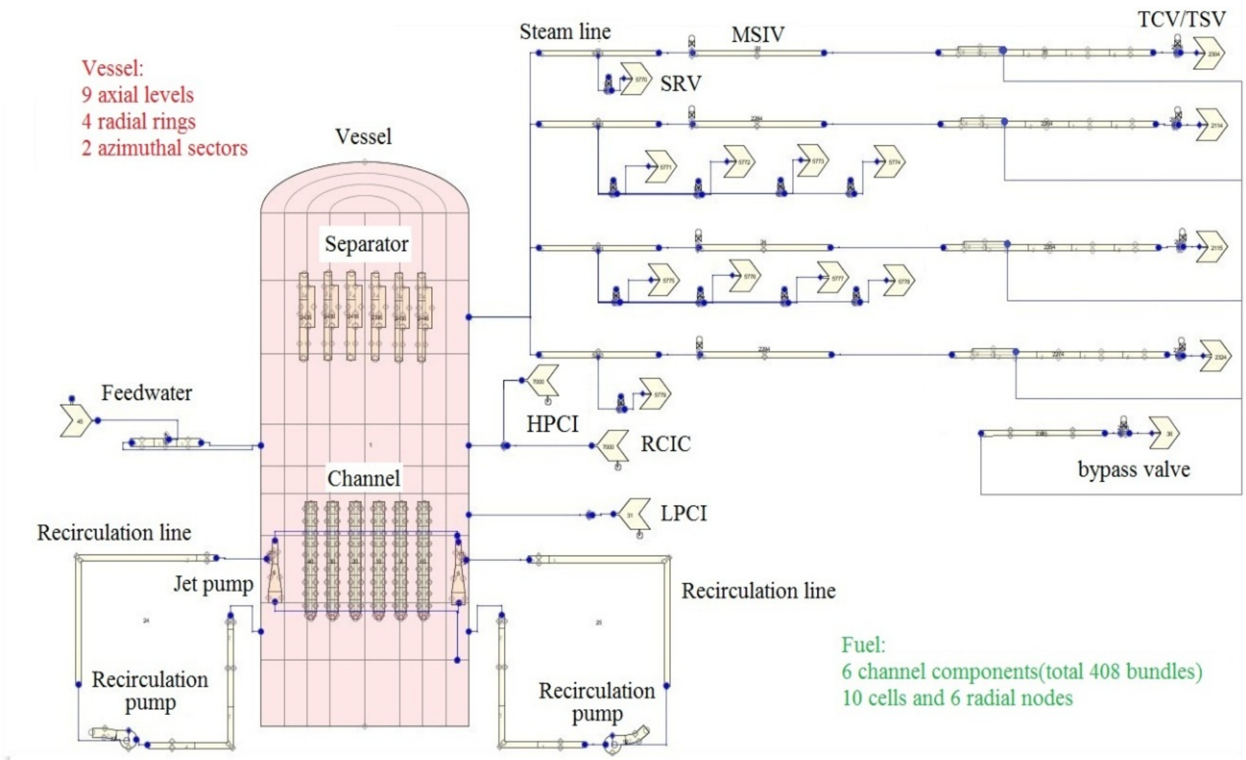

Figure 2 The TRACE/SNAP model of Chinshan NPP

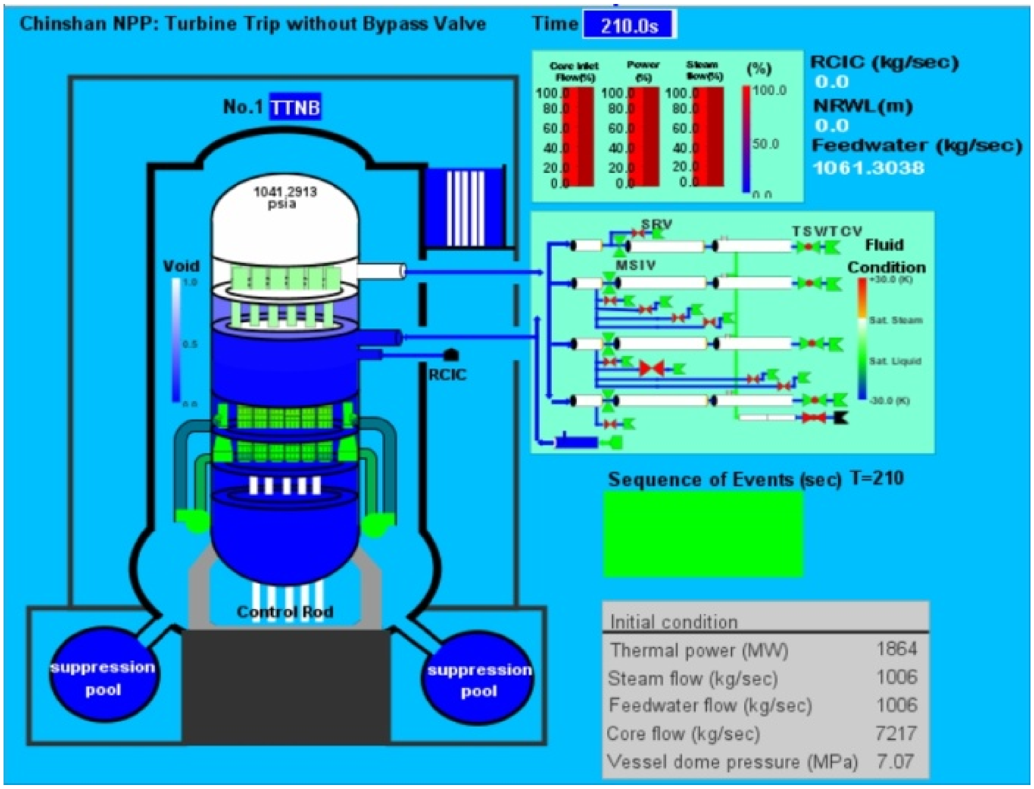

Figure 3 The animation model of Chinshan NPP

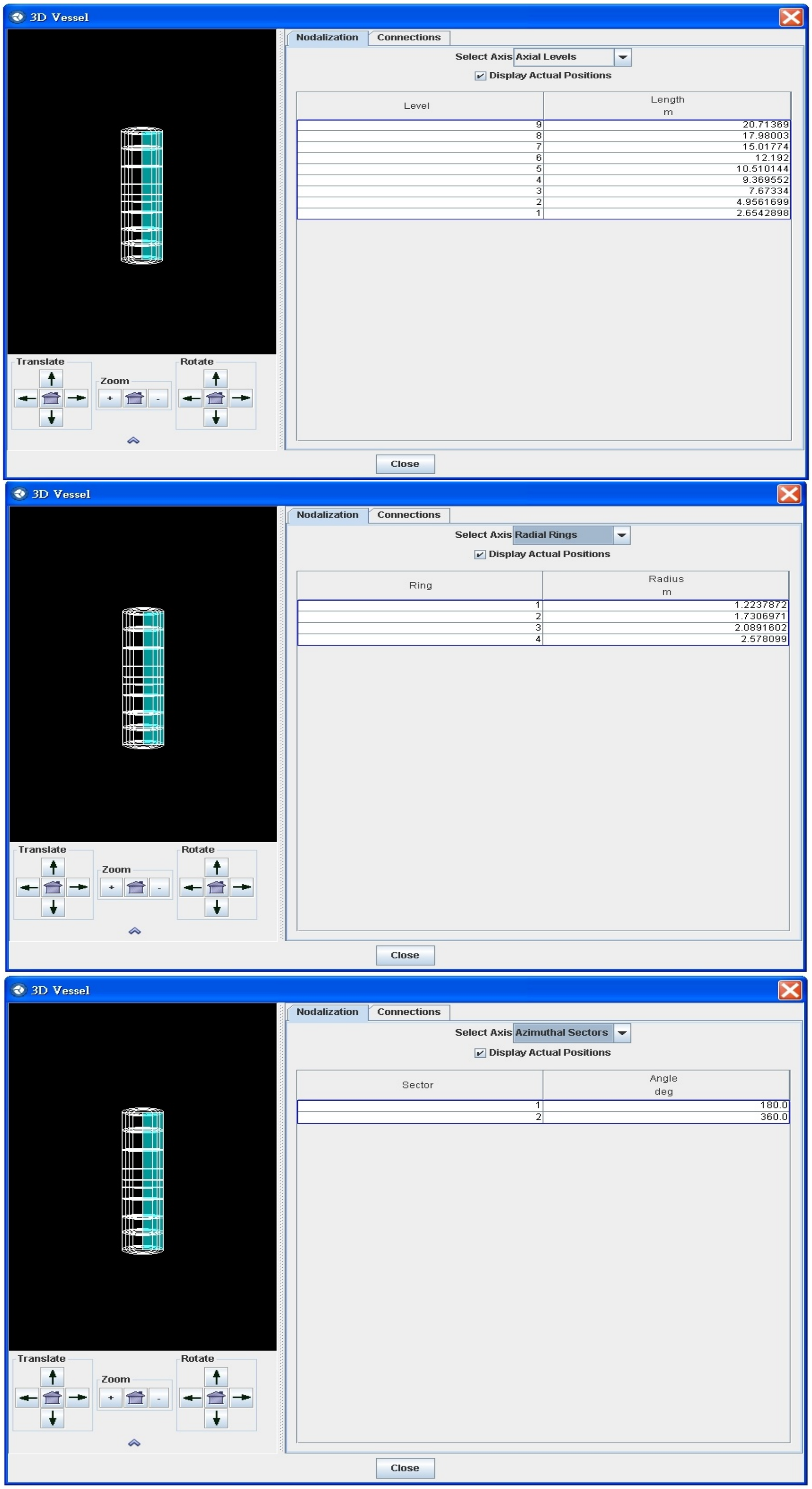

Figure 4 The 3D vessel condition of Chinshan NPP

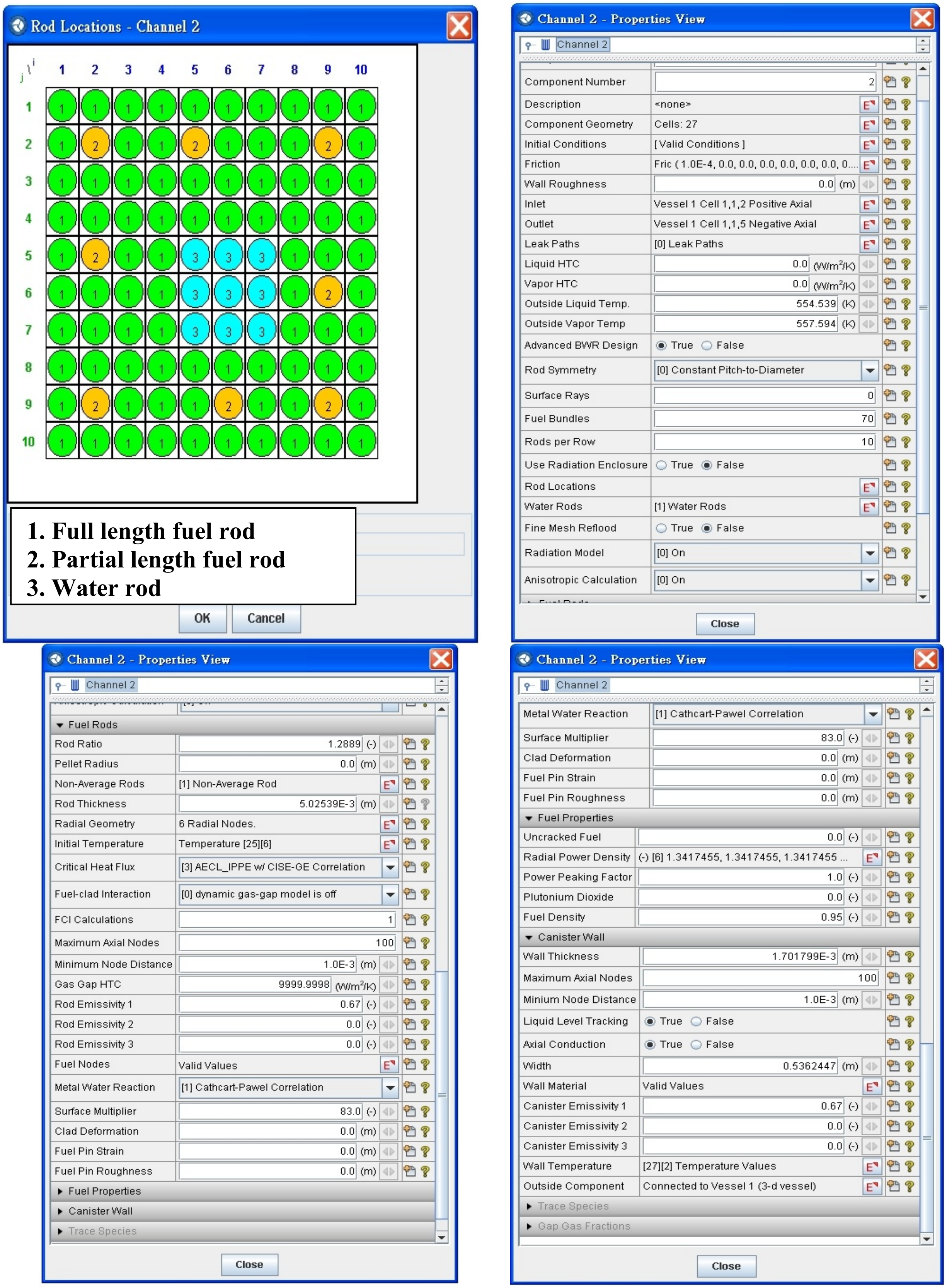

Figure 5 The channel condition of Chinshan NPP

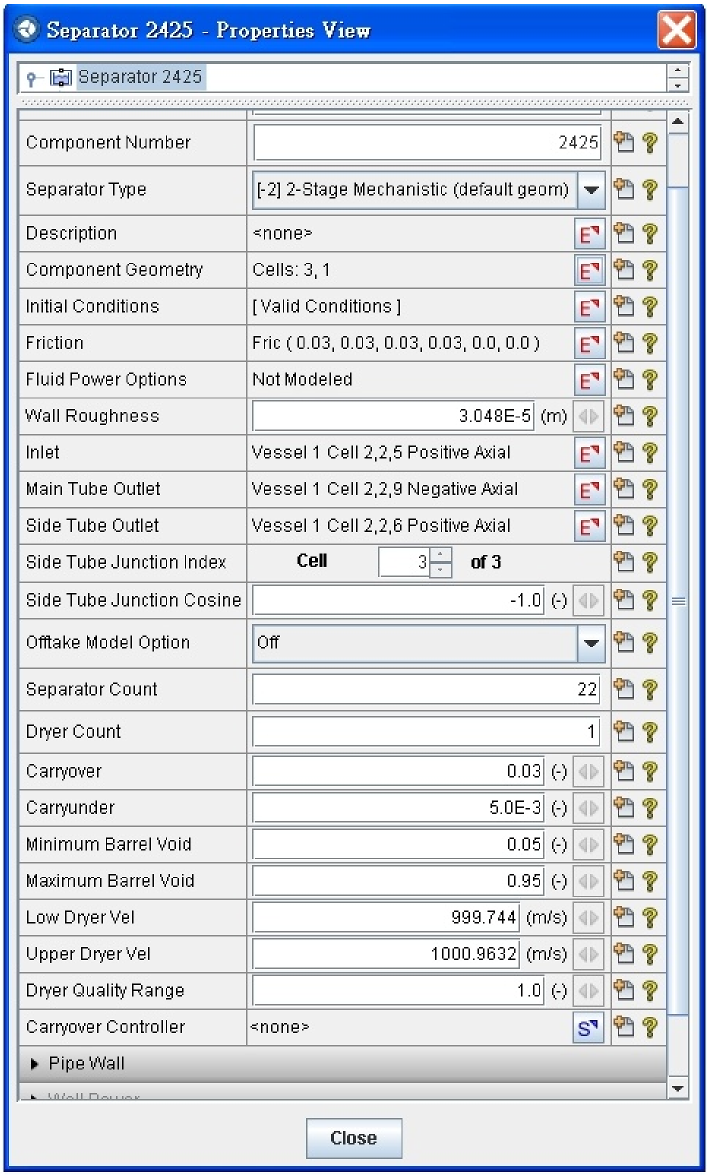

Figure 6 The separator condition of Chinshan NPP

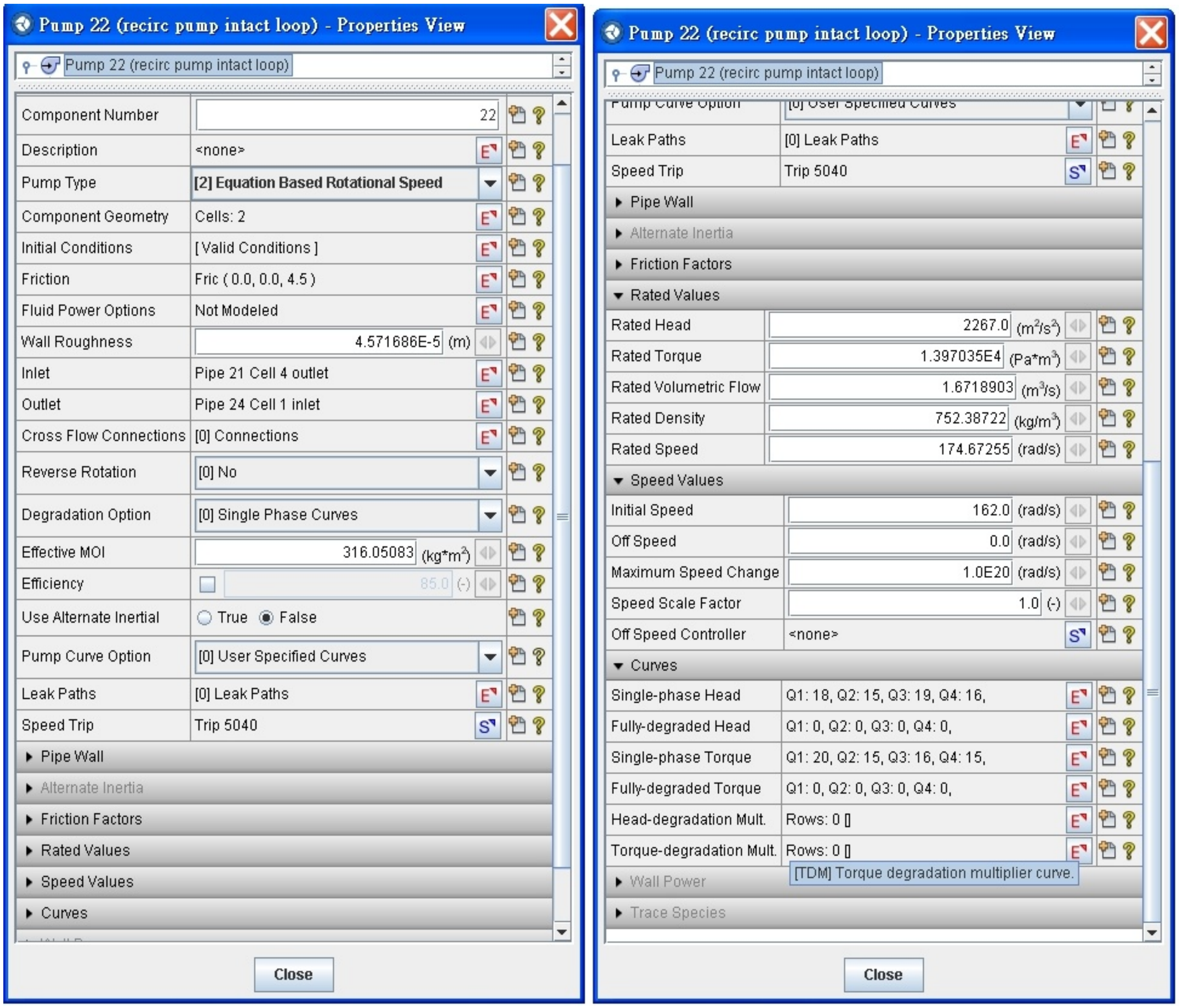

Figure 7 The recirculation pump condition of Chinshan NPP

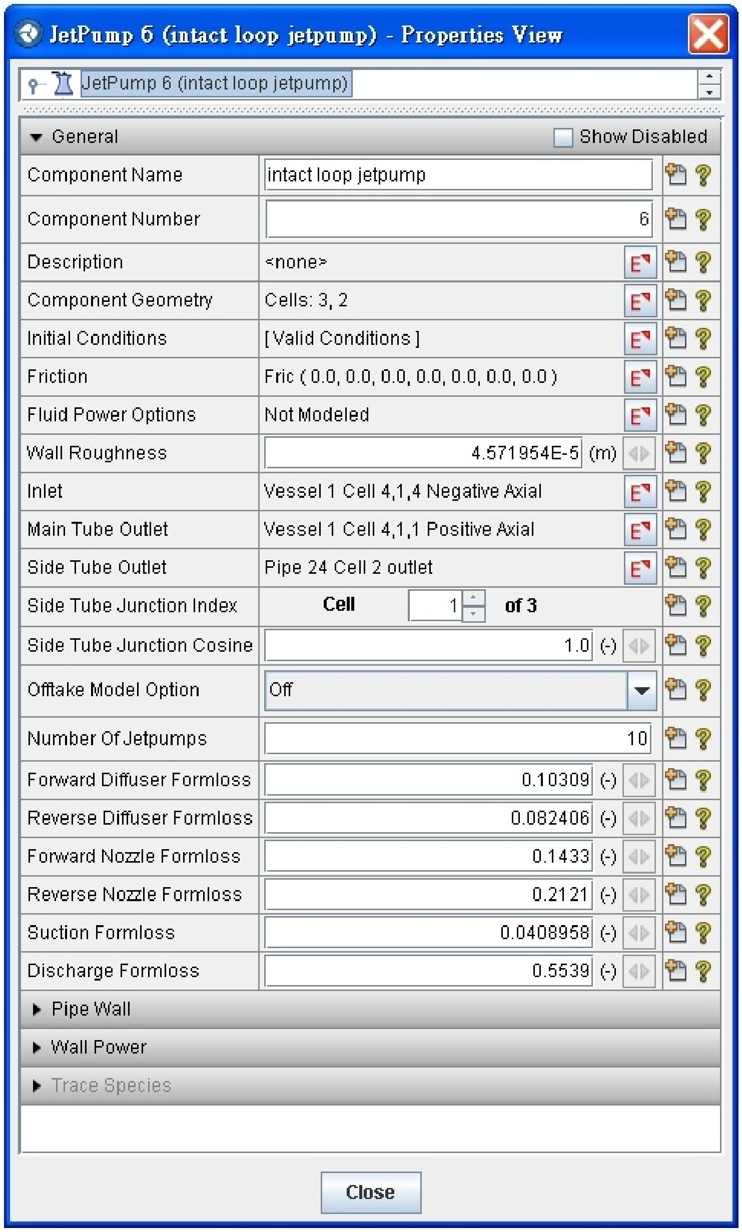

Figure 8 The jetpump condition of Chinshan NPP

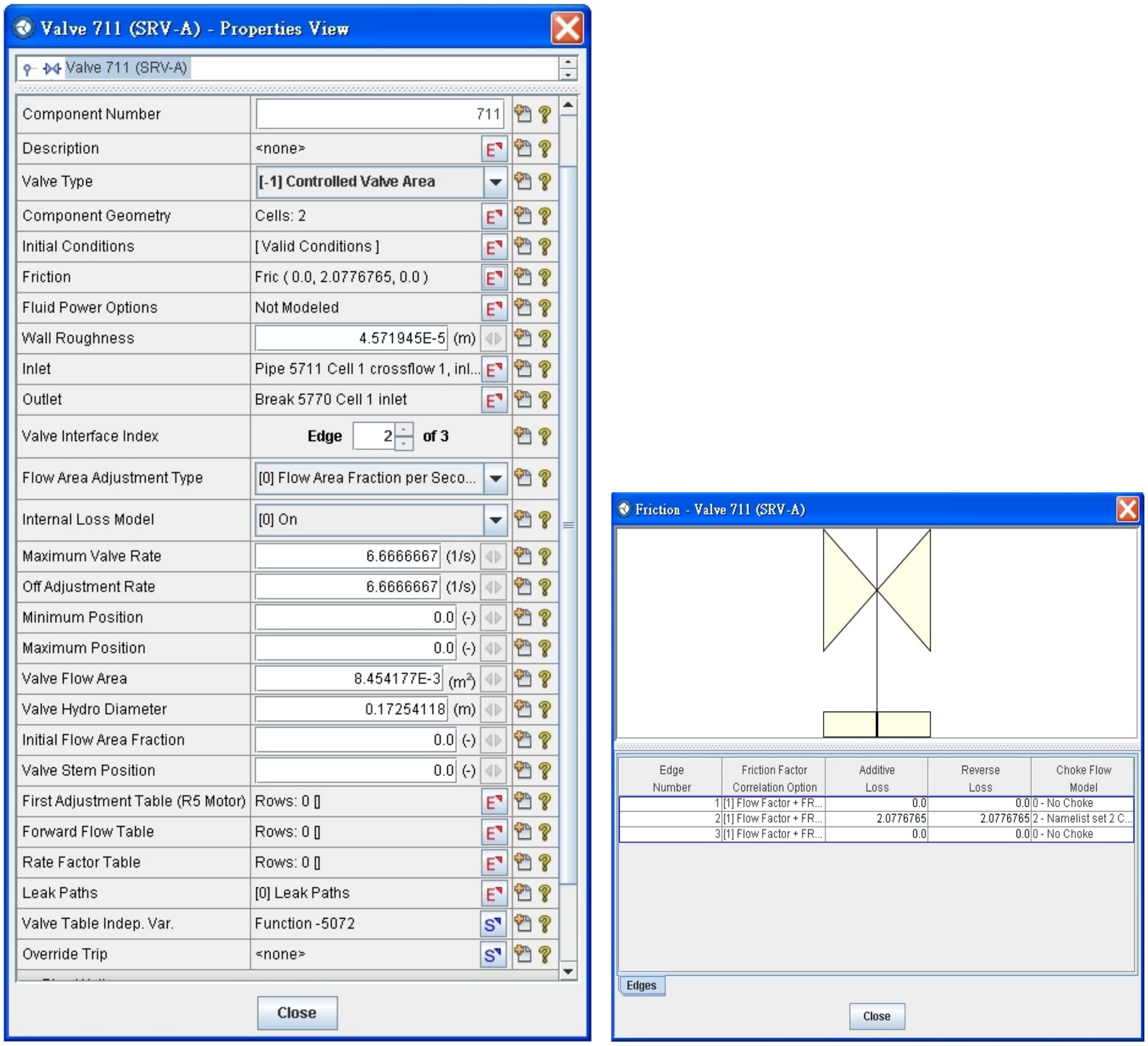

Figure 9 The SRV condition of Chinshan NPP

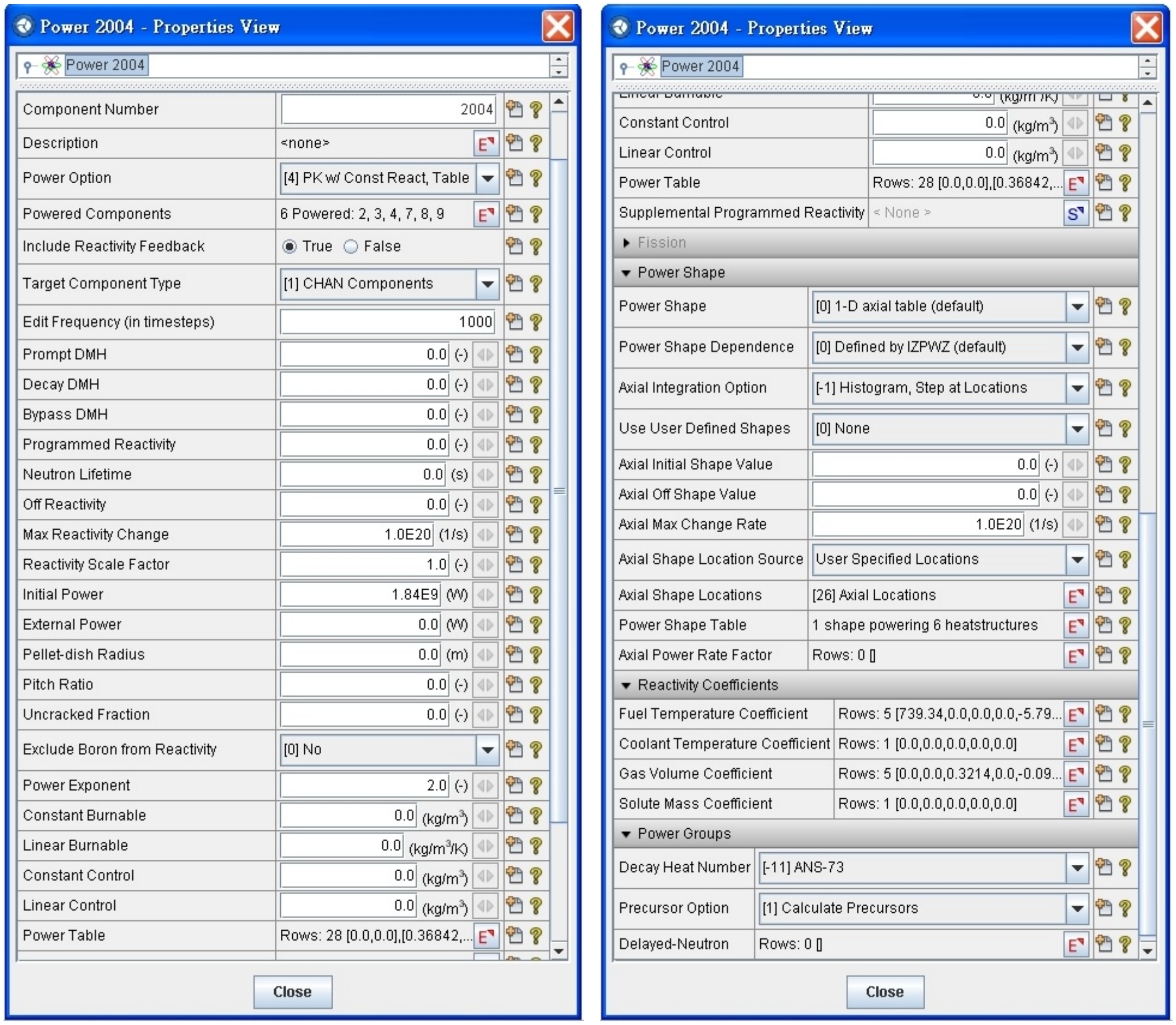

Figure 10　The power condition of Chinshan NPP

3. RESULTS

Before the transient calculation of Chinshan 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 and 2).

Table 1 The comparison of initial conditions between FSAR and TRACE data for turbine trip without bypass valve

Parameters	FSAR	TRACE	Difference (%)
Power (MWt)	1864	1864	0
Dome pressure (MPa)	7.13	7.07	-0.84
Core Inlet flow (kg/sec)	7240	7217	-0.32
Feedwater flow (kg/sec)	1009	1006	-0.3
Steam flow (kg/sec)	1009	1006	-0.3
NRWL (m)	1.10	1.11	0.91

Table 2 The comparison of initial conditions between FSAR and TRACE for MSIV closure

Parameters	FSAR	TRACE	Difference (%)
Power (MWt)	1864	1864	0
Dome pressure (MPa)	7.27	7.17	-1.38
Core Inlet flow (kg/sec)	7240	7217	-0.32
Feedwater flow (kg/sec)	1009	1006	-0.3
Steam flow (kg/sec)	1009	1006	-0.3
NRWL (m)	1.10	1.11	0.91

3.1 Turbine Trip without Bypass Valve Analysis

The reactor is initially operating at 105% of NBR power with vessel dome pressure of 1034.7 psi (7.13 MPa). Table 3 compares the turbine trip transient's sequences of FSAR data with TRACE. Their sequences are very similar. In turbine trip transient, the reactor scram and recirculation pump trip is initiated by position switches on the TSV when the valves are less than 90% open. The turbine bypass valve system is failed in this transient.

Fig.11~14 shows the results of TRACE. Fig. 11 depicts the power curves of FSAR data and TRACE. The trends of their curves are similar. The increase of the power was caused by the TSVs closing. The TSVs closing decreased the reactor's void fraction which generated the positive reactivity. Then, the scram initiated and the power dropped. Fig. 12 compares the steam dome pressure of FSAR and TRACE. The trends of the curves are approximately in agreement. The TSV closing caused the dome pressure to rise. Then, SRVs opened and led to the decline of dome pressure. Fig. 13 shows the comparison of core inlet flow between FSAR and TRACE. The curve of TRACE is consistent with the FSAR data. Due to the dome pressure increase, it resulted in the core inlet flow rising during 0.5~1.5 sec. Then, recirculation pumps trip caused the decrease of core inlet flow. In summary, the trends of TRACE prediction are consistent with FSAR data but there are a few differences in the values of the prediction. Because we cannot find the detailed FSAR analysis data, we don't know what the reasons cause the differences of TRACE results and FSAR data. Finally, by using TRACE results and SNAP's animation function, Fig. 14 shows the dome pressure and core inlet pressure of 3-D animation pictures. The higher dome pressure and core inlet pressure show in 2 sec and it can be observed from Fig. 14. Finally, Fig. 15 depicts the animation model of Chinshan NPP for this transient.

Table 3 The comparison of sequences between FSAR and TRACE data for turbine trip without bypass valve

Sequences	Time (s)	
	FSAR	TRACE
Transient started	0	0
TSV closure less than 90 percent open	0.015	0.015
TSV fully closure	0.1	0.1
Recirculation pump tripped	0.6	0.6
SRVs opened	0.95	1.04
Reactor scrammed	1	1
Reactor pressure rose to peak	2.4	2.45
End of the time	6	6

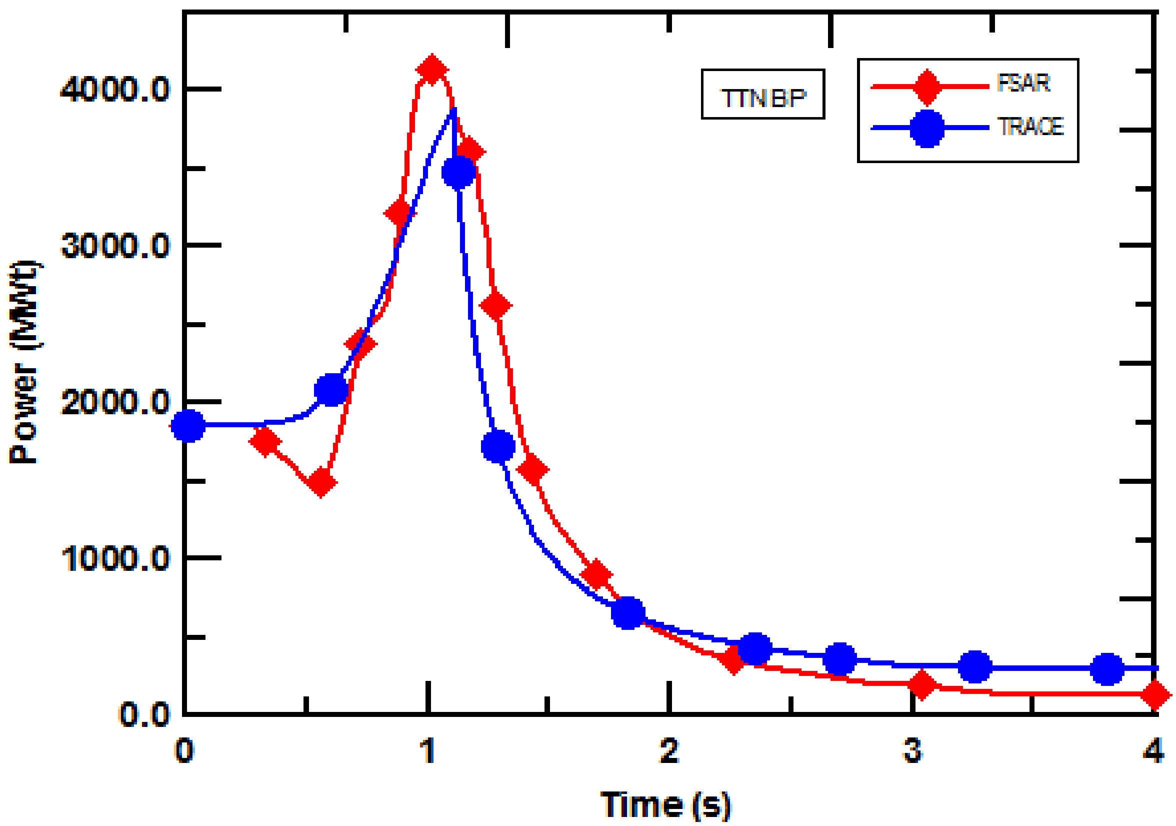

Figure 11 The comparison of power between FSAR and TRACE for turbine trip without bypass valve

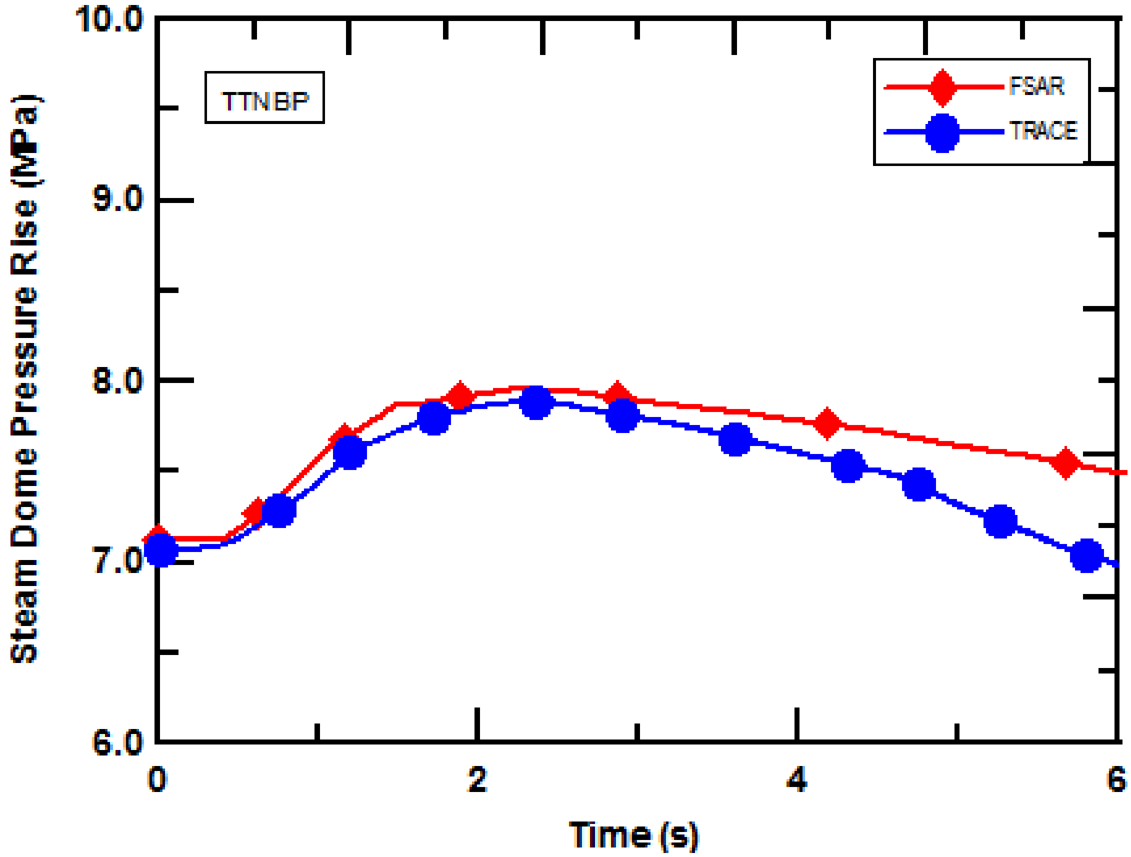

Figure 12 The comparison of steam dome pressure between FSAR and TRACE for turbine trip without bypass valve

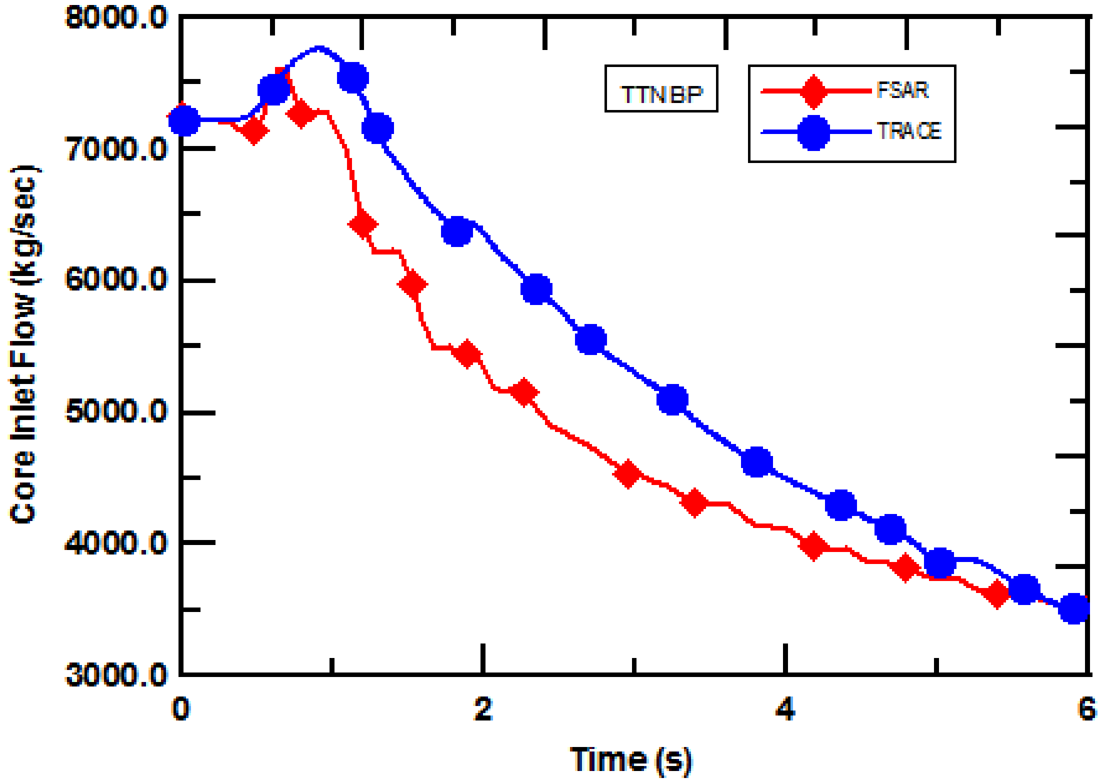

Figure 13 The comparison of core inlet flow between FSAR and TRACE for turbine trip without bypass valve

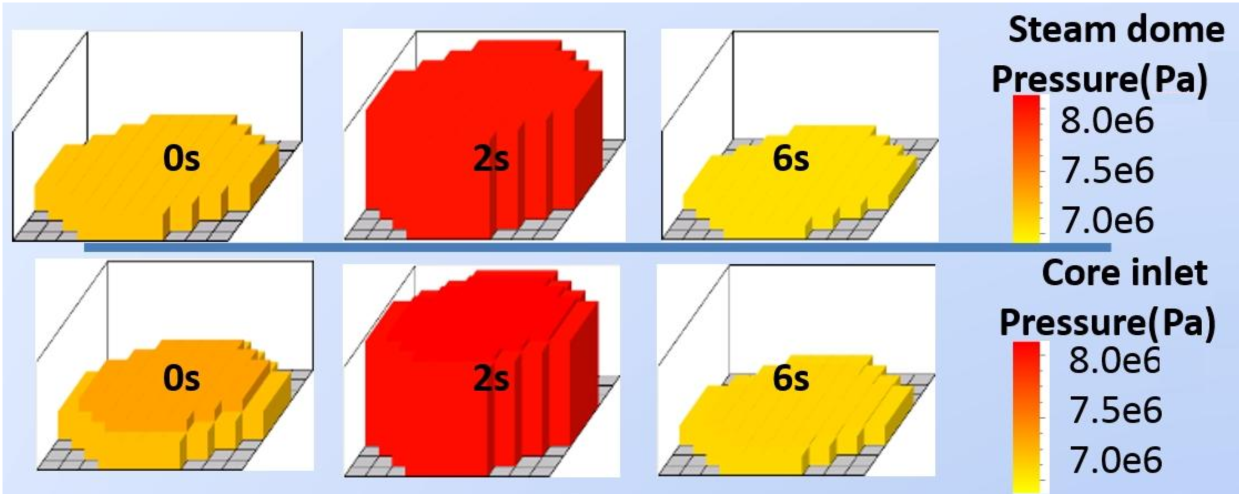

Figure 14 The 3-D picture of steam dome pressure and core inlet pressure for turbine trip without bypass valve

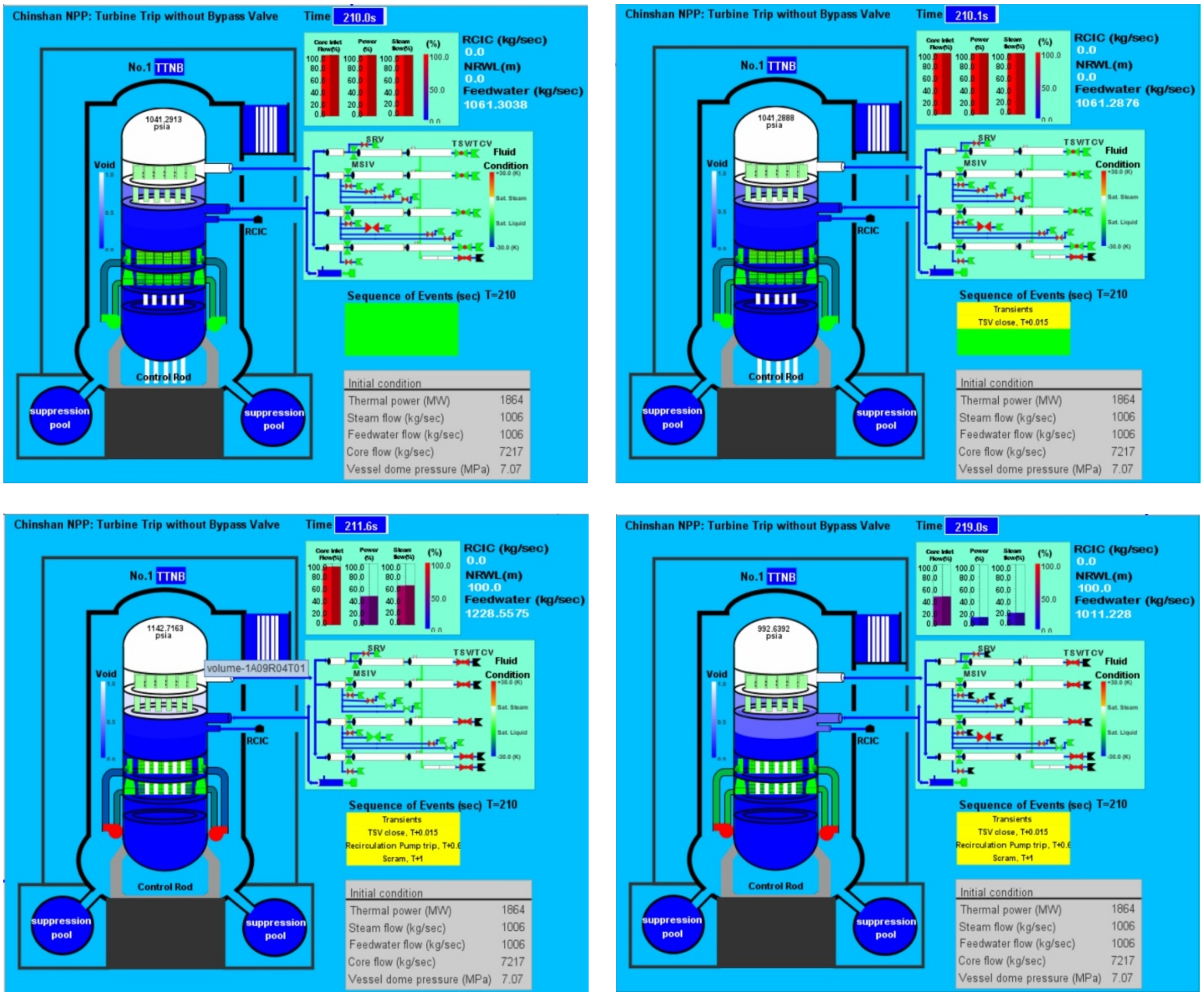

Figure 15 The animation model of Chinshan NPP for turbine trip without bypass valve

3.2 Main Steam Isolation Valve Closure Analysis

As main steam isolation valves close, position switches on the valves initiate a reactor scram when the valves in three or more main steamlines are less than 90 percent open. However, in the FSAR case, it assumes multiple failures of the position switch on each MSIV to initiate a reactor scram and reliance upon a neutron flux initiated scram. The reactor is initially operating at 105 percent of NB rated power with a vessel dome pressure of 1054.7 psi (7.27 MPa). Table 4 compares the MSIV closure transient's sequences of FSAR data with TRACE results. Their sequences are very similar. In MSIV closure transient, the reactor scram is initiated when the neutron flux rises to 122%. Then, the recirculation pumps trip and SRVs open.

Fig. 16~19 shows the comparisons of FSAR and TRACE. Fig. 16 depicts the power curves of FSAR and TRACE. The result of TRACE is consistent with FSAR data. The MSIVs closing resulted in the increase of the power. The MSIVs closing decreased the reactor's void fraction which generated the positive reactivity. Then, the scram was initiated by the neutron flux signal (122%) and the power started to decrease after 2 sec. However, at this time, the dome pressure still rose and the core inlet flow increased, which caused the void fraction still to decrease. It offered the positive reactivity which led to the power increase during 2.5~3 sec. Finally, the control rods fully inserted which resulted in the power dropped after 3 sec. Fig. 17 shows the steam dome pressure comparison of FSAR and TRACE. The trends of the curves are approximately in agreement. The steam dome pressure went up after the MSIV closure. When the dome pressure rose to the critical value, SRVs opened and led to the decline of dome pressure. Fig. 18 shows the core inlet flow results of FSAR and TRACE, and their curves are similar. The closure of MSIV caused the increase of the pressure. The increase of the pressure caused the core inlet flows rising during 1.5~3 sec. Then, the recirculation pump tripped which led to the decrease of core inlet flow. In summary, the TRACE results are consistent with FSAR data but there are a few differences in the values of the prediction. Due to the detailed FSAR analysis data cannot be found, we don't know what the reasons cause the differences of TRACE results and FSAR data. Finally, Fig. 19 shows the dome pressure and core inlet pressure of 3-D animation pictures by using TRACE results and SNAP's animation function. The higher pressures shows in 3.5 sec and it can be observed from Fig. 19. Finally, Fig. 20 depicts the animation model of Chinshan NPP for this transient.

Table 4 The comparison of sequences between FSAR and TRACE for MSIV closure

Sequences	Time (s)	
	FSAR	TRACE
MSIV initiated closure	0.0	0
Control rod ready to insertion (When Neutron flux rise to 122%)	1.5	1.37
SRVs opened	2.9	2.99
MSIV closure	3	3
Reactor pressure rose to peak	3.2	3.43
End of the time	7.5	7.5

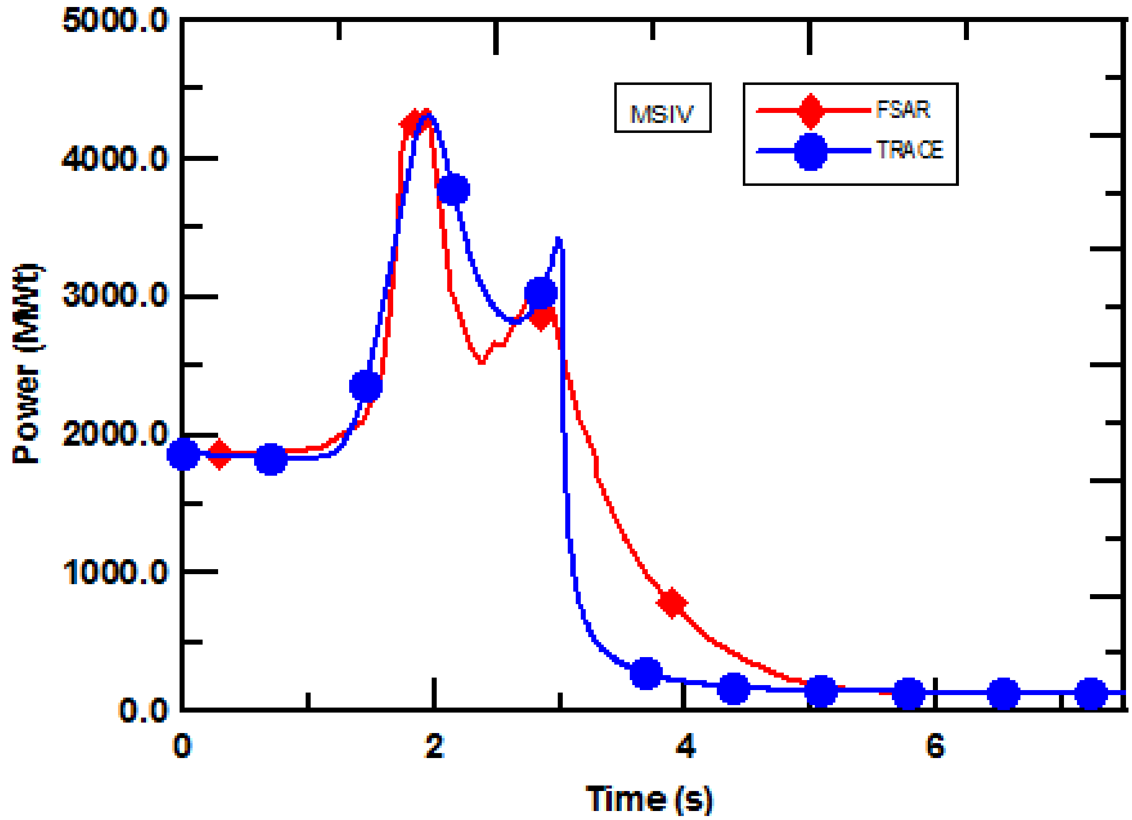

Figure 16 The comparison of power between FSAR and TRACE for MSIV closure

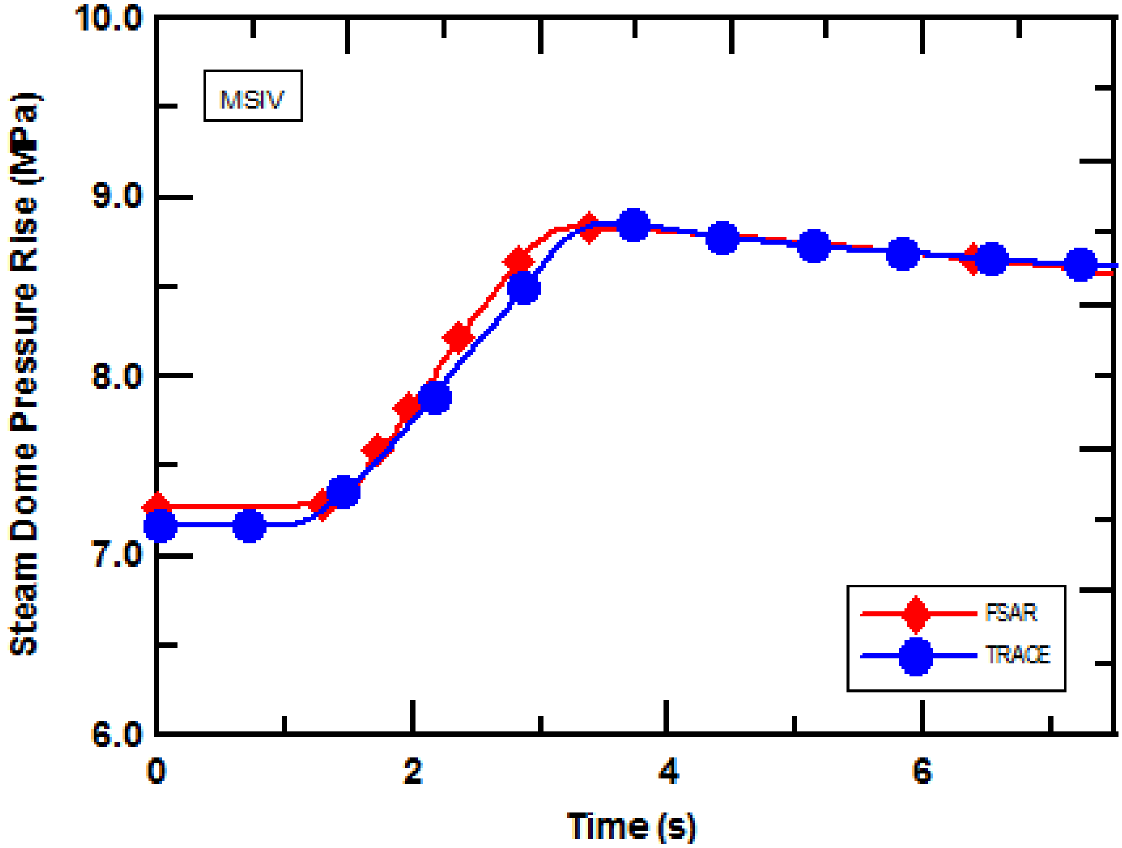

Figure 17 The comparison of steam dome pressure between FSAR and TRACE for MSIV closure

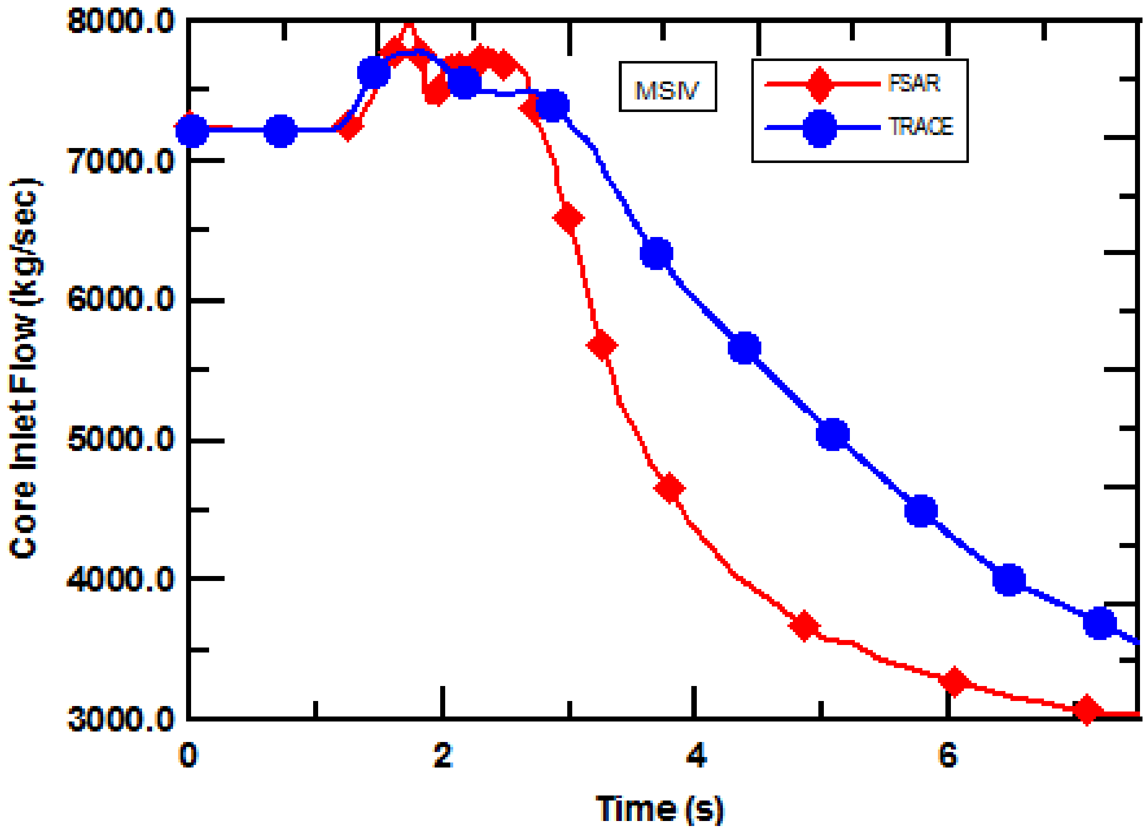

Figure 18 The comparison of core inlet flow between FSAR and TRACE for MSIV closure

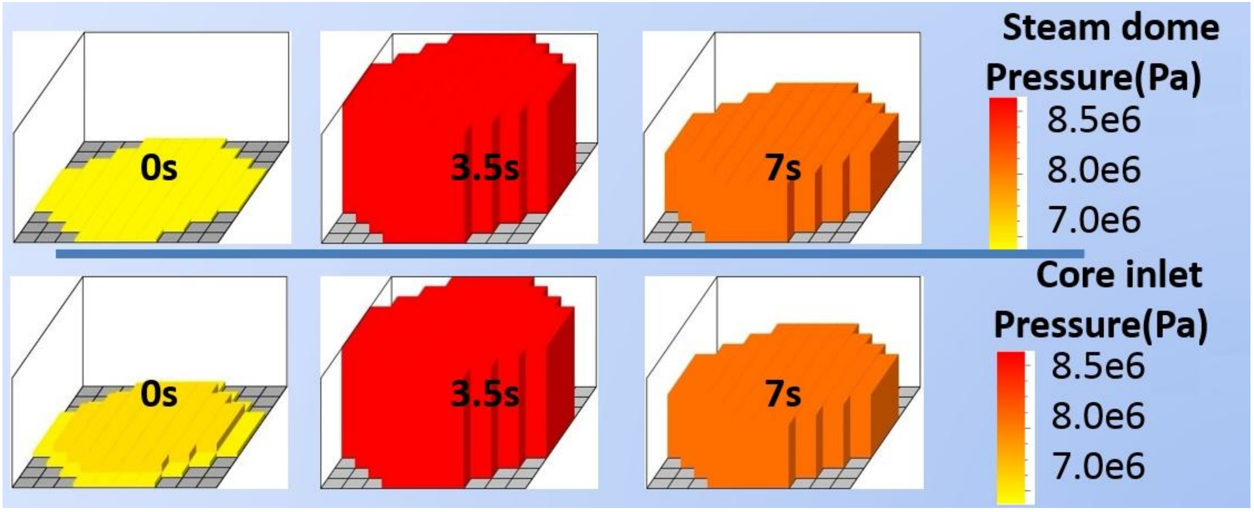

Figure 19 The 3-D picture of steam dome pressure and core inlet pressure for MSIV closure

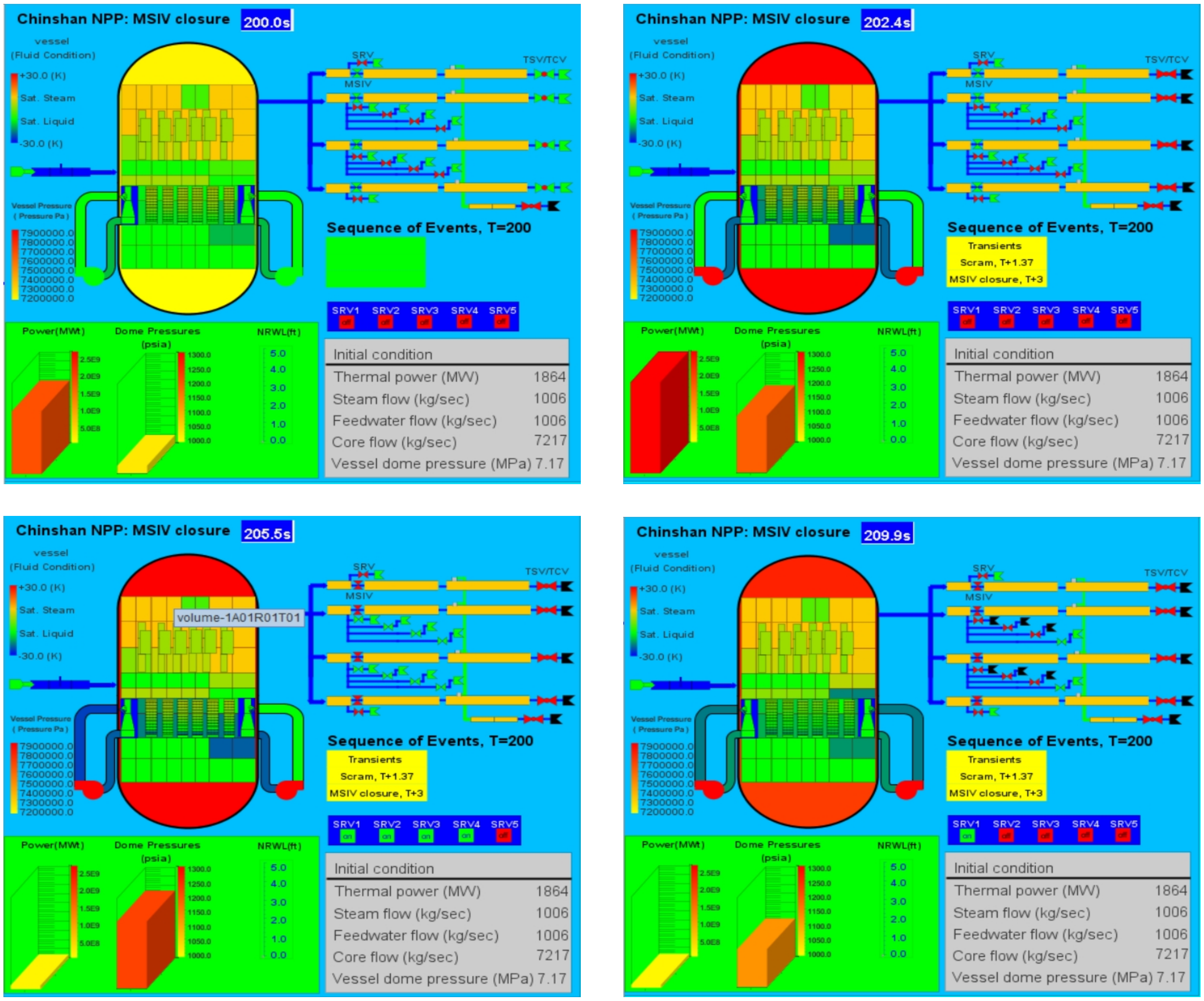

Figure 20 The animation model of Chinshan NPP for MSIV closure

3.3 Core Shroud Leakage Analysis

There are four assumed cases in the Chinshan 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 0.0254 m, the area of break is 0.07 m^2.
Case 2: the core shroud leakage occurred (in 2 azimuthal sectors of the vessel, 360 degree larger break), the length of break is 0.0254 m, the area of break is 11.1327 m^2.
Case 3: the case 2 condition + SBO + scram failed + RCIC failed.
Case 4: the case 2 condition + LOCA (one steamline break) + scram failed + RCIC failed.

The initial conditions of cases are 100% rated power/100% rated core flow and the sequences of case 1~4 are shown in Table 5~8. 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 transient (ex: recirculation pump, feedwater, turbine tripped) was performed for case 3 and the LOCA transient (ex: one steamline break, recirculation pump, feedwater tripped) was performed for case 4. Fig. 21~26 show the results of TRACE for all cases.

Fig. 21 and 22 show the power and Doppler reactivity results of TRACE. In all cases, when core shroud leakage occurred, the power increased rapidly (shown in Fig. 21). Then the power dropped sharply due to the void fraction increase. After core shroud leakage generated fully, the powers of case 1 and 2 reached the new steady state. However, case 3 and case 4 had more negative Doppler reactivity (shown in Fig. 22) due to the fuel temperature raise (shown in Fig. 25). So the power of case 3 and case 4 was lower than the power case 1 and case 2 after core shroud leakage generated fully.

Fig. 23 shows the water level results of Chinshan NPP core shroud leakage cases. In all cases, when core shroud leakage occurred, the water level rose rapidly and was larger than level 8. It also indicated that the level 8 signal was tripped. Therefore, if the Chinshan NPP finds the level 8 signal tripped, it may be the core shroud leakage happened. The oscillation happened for the case 1 and case 2 after the core shroud leakage occurred. Besides, the feedwater tripped and RCIC failed resulted in no water injection to the vessel. Due to no water supply, the water level of the case 3 and case 4 decreased continuously and were lower than the TAF (see Fig. 24). Due to one steamline break, the larger steam rushed out for case 4 that caused the water level lower than the BAF. The above results caused the fuel temperature increase for the case 3 and case 4 (shown in Fig. 25). The fuel temperature of the case 3 did not reach 1088K during 0~700 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 3 may be larger than 1088 K. Besides, because the fuel temperature of the case 4 was larger than 1088 K, the zirconium-water reaction happened (shown in Fig. 26). The above results indicated that the fuels might be damaged after the zirconium-water reaction happened. Finally, Fig. 27 depicts the animation model of Chinshan NPP for this transient.

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

	Time (sec)
Transient started	0
The core shroud leakage started to occur	200
NRWL reached L8	209
The core shroud leakage generated fully	210
Transient ended	700

Table 6 The sequences of case 2 for core shroud leakage transient

	Time (sec)
Transient started	0
The core shroud leakage started to occur	200
The core shroud leakage generated fully	201
NRWL reached L8	202
Transient ended	700

Table 7 The sequences of case 3 for core shroud leakage transient

	Time (sec)
Transient started	0
The core shroud leakage started to occur	200
SBO happened	200.1
Recirculation pump tripped	200.1
Turbine tripped	200.1
Feedwater tripped	200.1
The core shroud leakage generated fully	201
SRVs opened (first time)	202.02
NRWL reached L8	203.02
Core water level reached TAF	250.1
Transient ended	700

Table 8 The sequences of case 4 for core shroud leakage transient

	Time (sec)
Transient started	0
The core shroud leakage started to occur	200
One steamline broke	200.1
Feedwater tripped	200.1
The core shroud leakage generated fully	201
NRWL reached L8	201
Core water level reached TAF	204.02
Recirculation pump tripped	212.053
MSIV started to close	213.5
Core water level reached BAF	367.04
The temperature of cladding reached 1088K	377.04
Transient ended	700

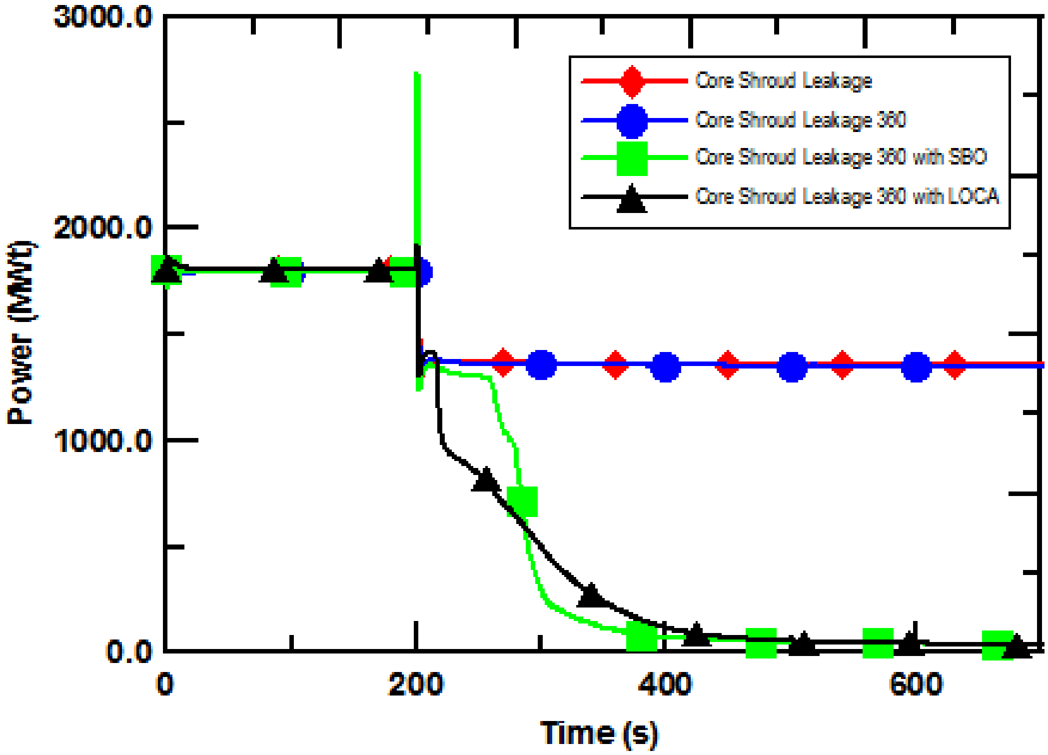

Figure 21 The TRACE results: power for core shroud leakage transient

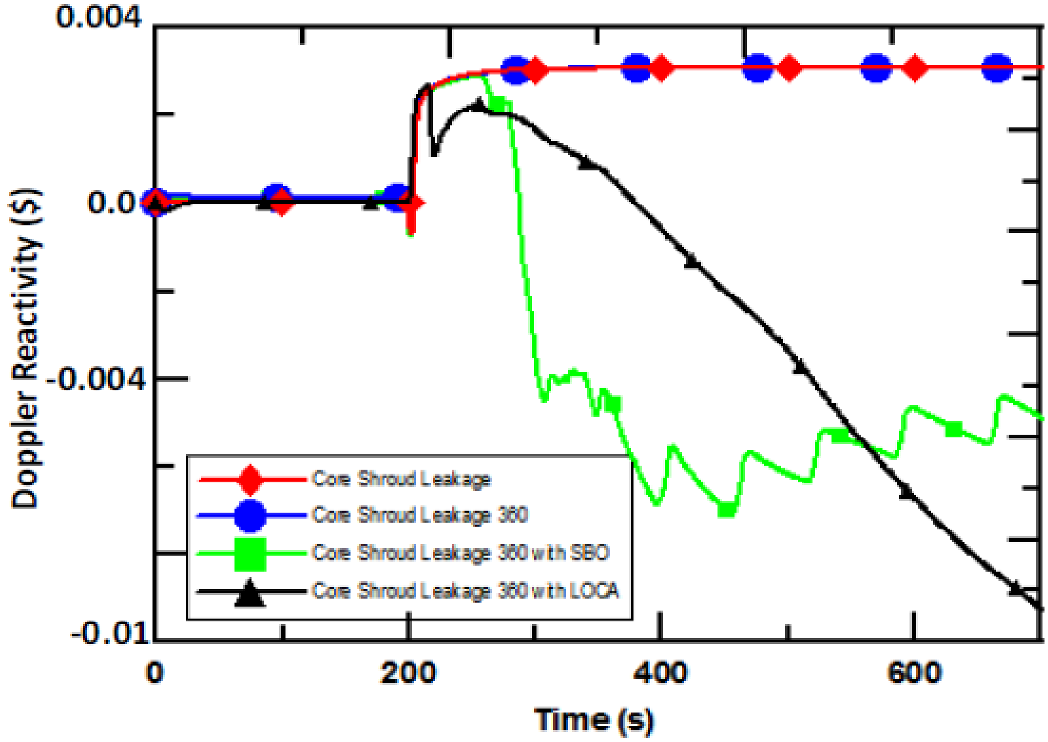

Figure 22 The TRACE results: Doppler reactivity for core shroud leakage transient

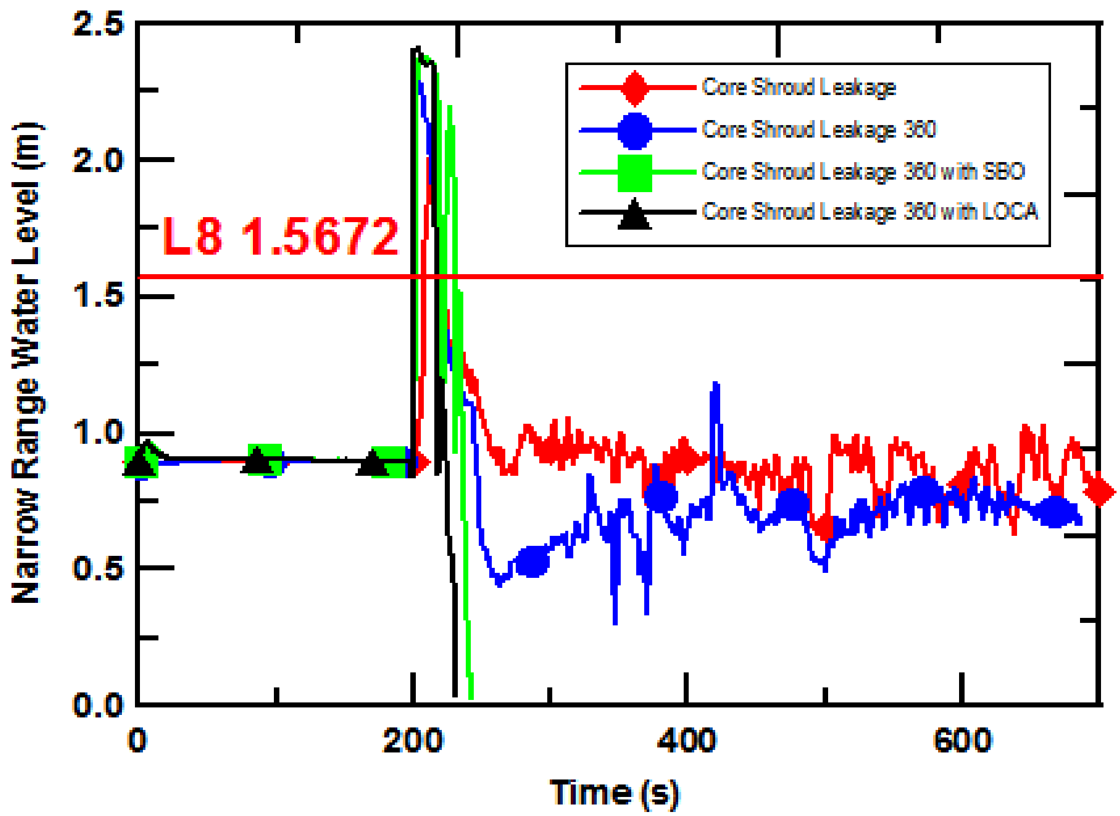

Figure 23 The TRACE results: NRWL for core shroud leakage transient

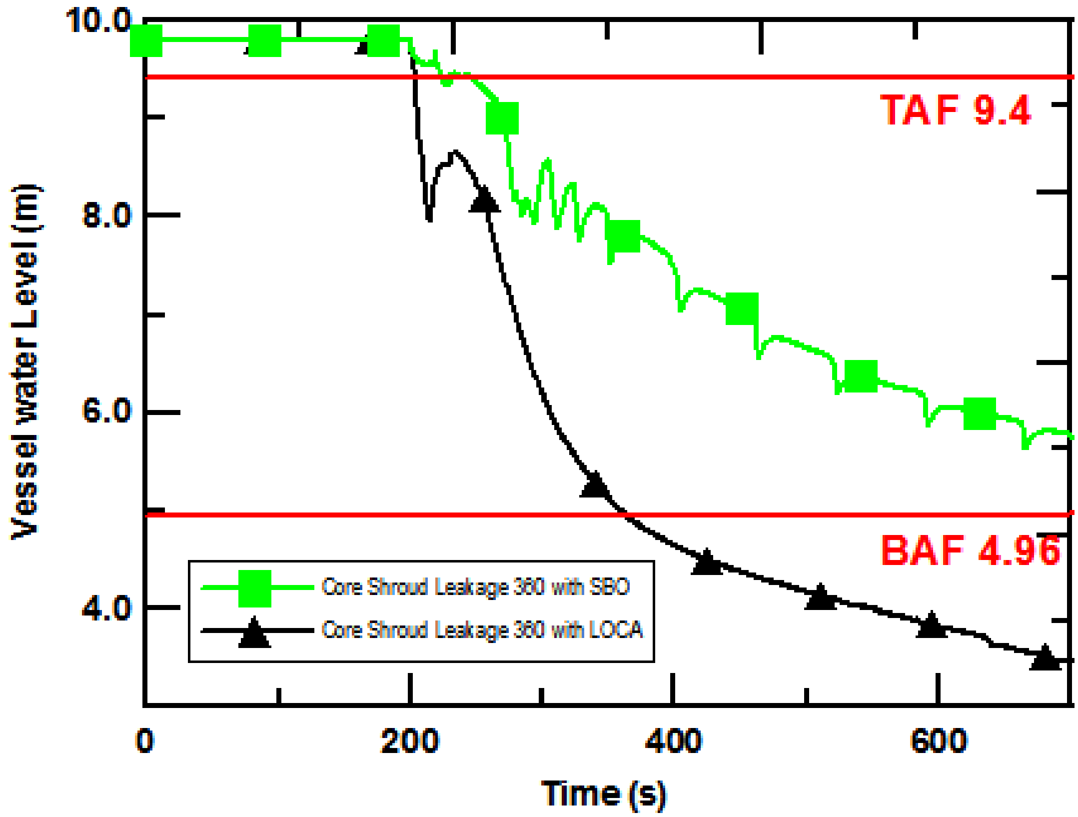

Figure 24 The TRACE results: vessel water level for core shroud leakage transient

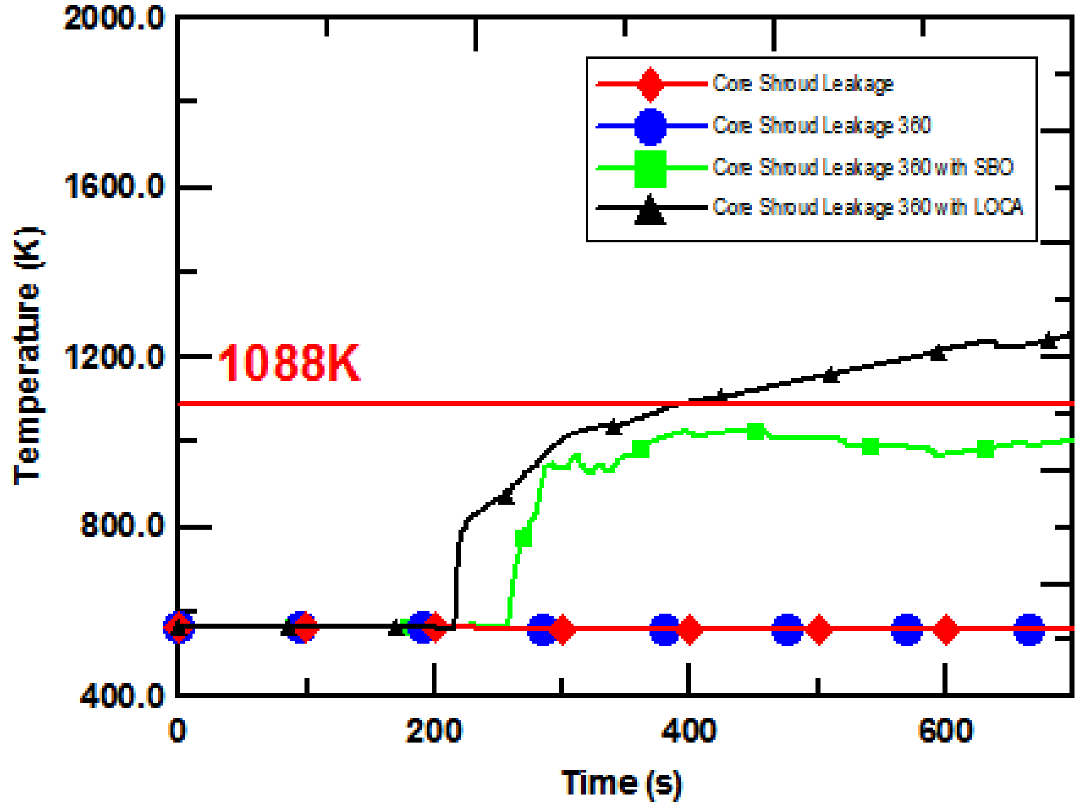

Figure 25 The TRACE results: cladding temperature for core shroud leakage transient

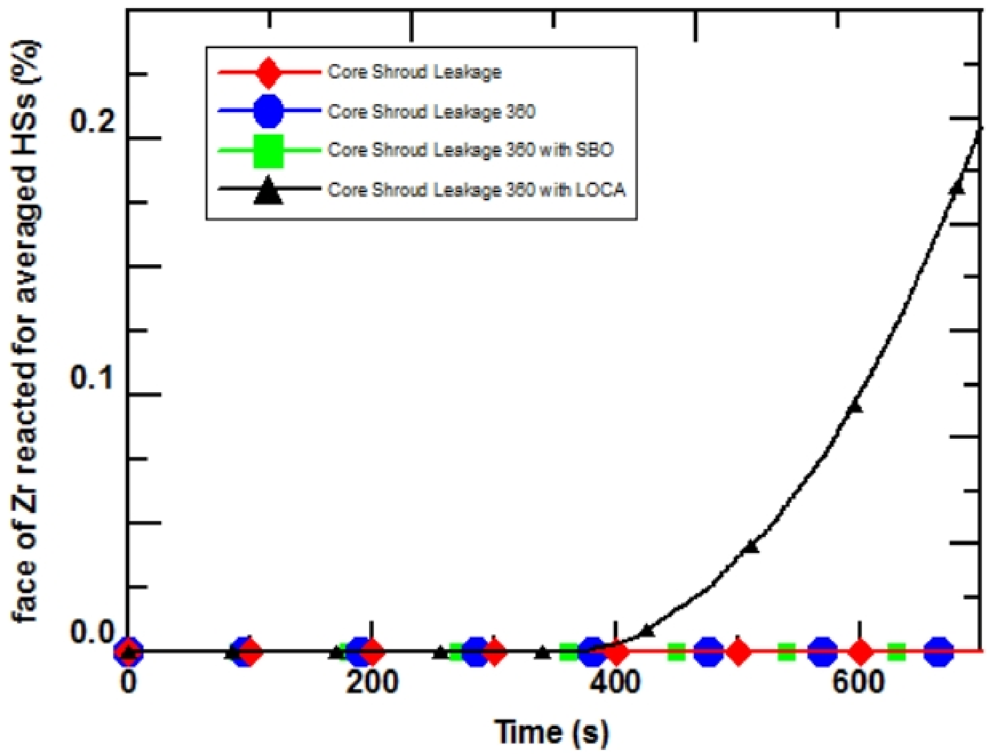

Figure 26 The TRACE results: cladding oxidation for core shroud leakage transient

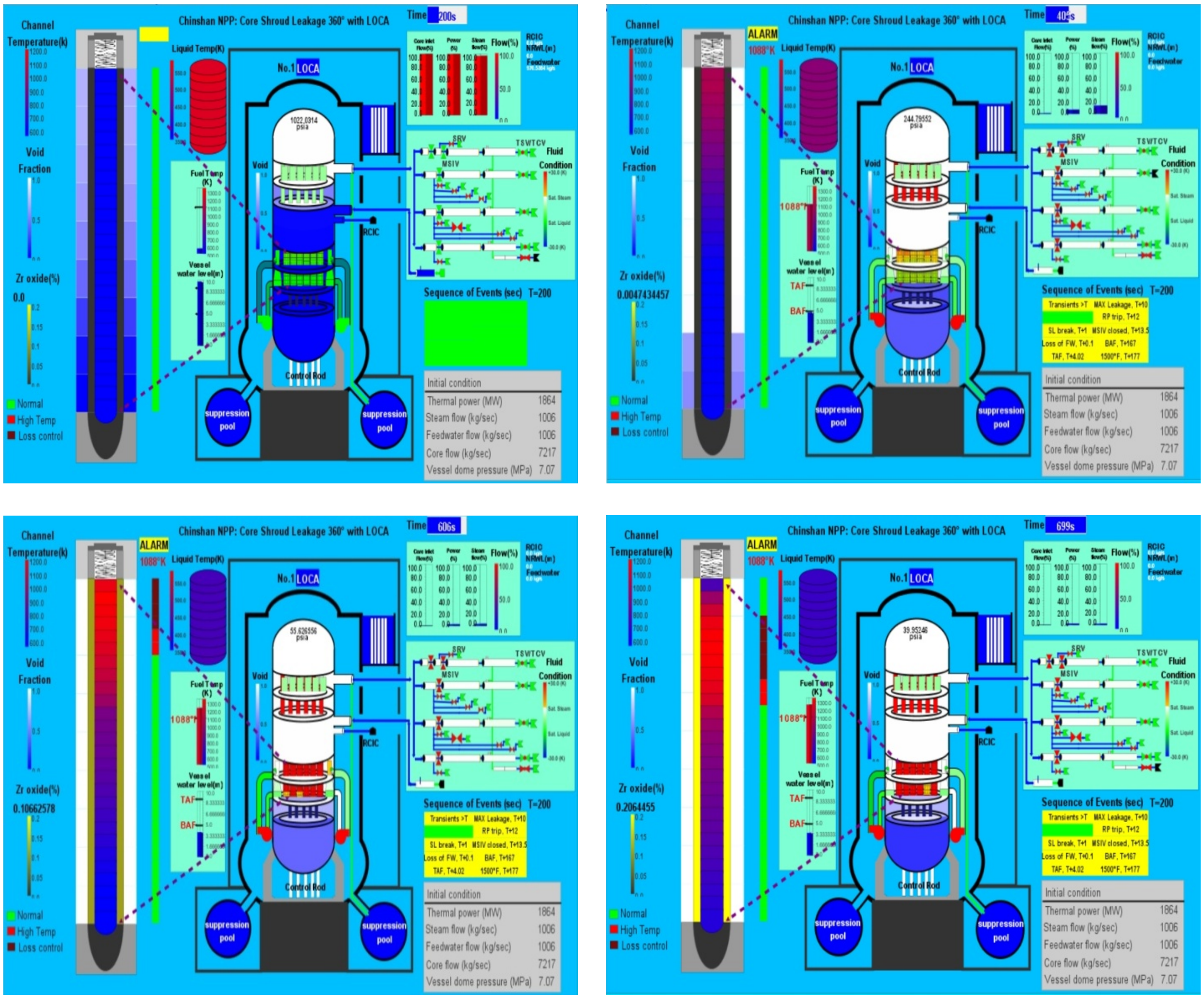

Figure 27 The animation model of Chinshan NPP for core shroud leakage transient

4. CONCLUSIONS

This research focuses on the development of the Chinshan NPP TRACE/SNAP model. The increase in reactor pressure transients including turbine trip and main steamline isolation valves closure were selected to validate the Chinshan NPP TRACE model. The results and sequences of TRACE are similar to the FSAR data for the turbine trip and main steamline isolation valves closure transients. Especially in the main steamline isolation valves closure transient, it shows the highest steam dome pressure happened (about rise 8.8 MPa) and the systems responses of TRACE are nearly the same with the FSAR data. By the above compared results, it indicates that there is a respectable accuracy in the Chinshan NPP TRACE/SNAP model and it also shows that the Chinshan NPP TRACE/SNAP model is satisfying for the purpose of Chinshan NPP safety analyses with confidence.

This study also developed the TRACE/SNAP core shroud leakage models of Chinshan NPP. There are four 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 Chinshan 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 and 2) wasn't influence the Chinshan NPP safety. However, the core shroud leakage + station blackout (SBO) or one steamline break (LOCA) transient (case 3 and 4) may cause the cladding temperature larger than 1088 K and influence the Chinshan NPP safety.

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10. SUPPLEMENTARY NOTES
K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)
Chinshan Nuclear Power Plant (NPP) is the first NPP in Taiwan which is of the BWR/4 type plant. This research focuses on the development of the Chinshan NPP TRACE/SNAP model. In order to check the system response of the Chinshan NPP TRACE/SNAP model, this study uses the analysis results of Final Safety Analysis Report (FSAR) to assess the Chinshan NPP TRACE/SNAP model. The increase in reactor pressure transients including turbine trip and main steam isolation valves closure were selected to validate the Chinshan NPP TRACE/SNAP model. The trends of TRACE analysis results were consistent with the FSAR data. It indicates that there is a credible fidelity in the Chinshan NPP TRACE/SNAP model. In addition, this research also investigates the application of the Chinshan NPP TRACE/SNAP model for the core shroud leakage. The core shroud leakage is one of current issues of concern of the U.S. NRC and other BWR/4 NPP owners. This research utilizes the Chinshan NPP TRACE/SNAP model to perform the core shroud leakage transients for Chinshan NPP safety analysis. The TRACE analysis results show that the simple core shroud leakage transient does not influence the Chinshan NPP fuel temperature to cause any core damage. However, the core shroud leakage combined with station blackout (SBO) or steamline break (loss of coolant accident, LOCA) transient would cause the cladding temperature to reach higher than 1088 K and affected the Chinshan NPP safety.

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