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International Agreement Report

Assessment of LONF ATWS for Maanshan PWR Using TRACE Code

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ABSTRACT

The objective of this study is to utilize TRACE code to analyze the reactor coolant system (RCS) pressure transients under LONF (Loss of Normal Feedwater) ATWS (Anticipated Transient Without Scram) for Maanshan PWR. TRACE is an advanced thermal hydraulic code for nuclear power plant safety analysis which is developed by U.S. NRC. Maanshan nuclear power plant (NPP) is a Westinghouse three-loop PWR in Taiwan. The rated core thermal power of Maanshan with MUR-PU (Measurement Uncertainty Recapture Power Uprate) is 2822 MWt. According to Westinghouse anticipated transients without trip report, LONF ATWS was regarded as the most severe plant condition. The ASME Code Level C service limit criteria of 22.06 MPa (3200 psig) was assumed to be an unacceptable plant condition in SECY-83-293.

In order to conform to 10 CFR 50.62, Maanshan NPP has set up AMSAC that is diverse from reactor trip system to automatically initiate the auxiliary feedwater system and initiate a turbine trip under conditions indicative of an ATWS. Since the ATWS analysis is not specified in the FSAR, we use TRACE code to assess the RCS pressure for Maanshan NPP. The results indicate that RCS pressure could keep within 22.06 MPa with sufficient negative moderator temperature coefficient (MTC) and normal work of AMSAC and valves.

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 ATWS model of Maanshan NPP has been built. In this report, we focus on the TRACE analysis of LONF ATWS.

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EXECUTIVE SUMMARY

An agreement which includes the development and maintenance of TRACE has been signed between Taiwan and USA on CAMP. INER (Institute of Nuclear Energy Research Atomic Energy Council, R.O.C.) 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 model of Maanshan NPP is developed.

Maanshan NPP is the first PWR in Taiwan. Its reactor is made by Westinghouse Company and has the rated power of 2822 MWt. The reactor coolant system has three loops and each loop has a reactor coolant motor and a steam generator. Besides, the pressurizer is connected with the hot-leg piping in loop 2.

The codes used in this research are TRACE v5.0p3 and SNAP v2.2.1. The Maanshan PWR TRACE model is based on Wang et al. [1][2] that V. & V. with FSAR [3] and start-up tests. In order to simulate the conditions of ATWS, we establish PORVs, SVs, spray system, AFW system, and AMSAC setting etc. into the model. Before transient simulation, it is necessary for testing the convergence of steady state of the Maanshan NPP TRACE model and comparing the TRACE data in a steady state. After completing the components in TRACE model, then introduce the AMSAC setting against Westinghouse Anticipated transients without trip report. Furthermore, we make some sensitivity studies as MTC variations, RCP trip, and failure of partial motor-driven AFW. The results indicate that RCS pressure could keep within 22.06 MPa with sufficient negative MTC and normal work of AMSAC and valves.

ABBREVIATIONS

AFW	Auxiliary Feedwater
AMSAC	ATWS Mitigation System Actuation Circuitry
ATWS	Anticipated Transient Without Scram
CAMP	Code Applications and Maintenance Program
CST	Condensate Storage Tank
DST	Demineralized Water Storage Tank
ESFAS	Engineered Safety Features Actuation System
FSAR	Final Safety Analysis Report
INER	Institute of Nuclear Energy Research Atomic Energy Council, R.O.C.
LONF	Loss of Normal Feedwater
MTC	Moderator Temperature Coefficient
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
PORVs	Power-Operated Relief Valves
RCS	Reactor Coolant System
RPS	Reactor Protection System
RTS	Reactor Trip System
S/G	Steam Generator
SNAP	Symbolic Nuclear Analysis Program
SVs	Safety Valves
TAF	Top of Active Fuel
TPC	Taiwan Power Company
TRACE	TRAC/RELAP Advanced Computational Engine
US	United States

1. INTRODUCTION

Maanshan nuclear power plant is the only Westinghouse PWR of Taiwan Power Company (Taipower, TPC). A few years ago, TPC has made many assessments in order to uprate the power of Maanshan NPP [4]. The assessments include NSSS (Nuclear Steam Supply System) parameters calculation, uncertainty acceptance, integrity of pressure vessel, reliability of auxiliary systems, and transient analyses, etc. Maanshan NPP finally uprates to 2822 MWt since 2009.
The USNRC is developing an advanced thermal hydraulic code named TRACE for nuclear power plant safety analysis. The development of TRACE is based on TRAC and integrating with RELAP5 and other programs. SNAP is an integrated suite of programs designed to provide an efficient framework for the user of nuclear safety analysis codes. Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program) which includes the development and maintenance of TRACE. Since Taiwan participates in the activities of CAMP, we have to re-analyze these transients with TRACE/SNAP to confirm their credibility. ATWS for Maanshan with MUR is one of the tasks.

The LONF ATWS results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the turbine fails to trip immediately, the secondary water inventory will decrease significantly before the actuation of AFW system. The heat removal from the primary side decreases, and this leads to increases of primary coolant temperature and pressure. The water level of pressurizer also increases subsequently. The heat removal through the relief valves and the auxiliary feedwater is not sufficient to fully cope with the heat generation from primary side. The pressurizer will be filled with water finally, and the RCS pressure might rise above the set point of relief valves for water discharge. Then the transient proceeds by the negative reactivity feedback due to the temperature increase of coolant. The RCS pressure may reach its peak after core power reduction [5].

The peak RCS pressure depends on steam generator inventory, primary coolant temperature, negative reactivity feedback, and core power, etc.

2. DESCRIPTION OF MAANSHAN TRACE ATWS MODEL

The code versions adopted in this study are TRACE v5.0p3 and SNAP v2.2.1. The Maanshan PWR TRACE model is based on Wang et al. [1][2] that verified with start-up test. It is a three-loop model, the main components include the reactor pressure vessel, pressurizer, steam generators, steam piping in the secondary side (including four sets of steam dump and vent valves) and the steam dump control system. Figure 1 shows the whole TRACE model for Maanshan NPP, and individual description below.

2.1 Reactor Pressure Vessel

The vessel in TRACE is a unique model with three-dimensional geometry rather than one-dimensional that most conventional simulation tools are. We define the vessel model for Maanshan as 12 levels in the axial direction, two rings in the radial direction (internal and external rings) and six equal azimuthal sectors. The control-rod conduit connects the 12th and 7th layer of the vessel from end to end. The fuel region is between the third and sixth layer, and heat conductors are added onto the structures to simulate the reactor core. Each volume has individual thermal-hydraulic parameters to assist us in refining the core. Its divisions are shown in Figure 2.

2.2 Pressurizer

The pressurizer is indispensable in ATWS transients for the purpose of pressure regulation at primary side. It is a vertical and cylindrical tank with carbon steel, and the components include heater, spray, power-operated relief valves, and safety valves, etc. Since the pressurizer plays an essential role in a PWR, its design parameters listed in Table 1 must be conformed to the plant data.

2.3 Steam Generator with Feedwater Control System

The steam generator of Maanshan NPP is a Model F, vertical U-tube heat exchanger, with a total of 5624 U-tubes. Figure 3 shows the TRACE model of the steam generator, and the U-Tube in the primary side is divided into 18 volumes. A FILL component represents ‘‘Hot-leg fluid inflow,” and a BREAK component is used to represent ‘‘Cold-leg fluid outflow.” Their inputs were derived from real plant temperature and pressure time histories [6][7]. On the secondary side, the division of volume is seven for boiler, 13 for downcomer, 13 for steam dome and separator. Furthermore, a FILL component is added to represent ‘‘Feedwater inflow,” and a BREAK component is added to represent ‘‘steam outflow.” Plant data for feedwater flow and other input parameters derived from velocity. Temperature and pressure are used to set initial conditions. Feedwater flow is controlled by a three-variable feedwater control system after the transient began.

2.4 Steam Dump Control System

The steam dump control system is composed of ten atmospheric venting valves, six turbine bypass valves and the associated piping control apparatus. Figure 4 shows the TRACE model of the steam dump control system. This model was established mainly as described in the report of INER [8]. The ten atmospheric venting valves and six turbine by-pass valves are grouped into four sets in this model: three turbine bypass valves comprise the first set; the other three are formed as the second set; five atmospheric venting valves are considered as the third set, and the fourth set consists of the rest.

2.5 PORVs and SVs of Pressurizer and Main Steam Line

In order to regulate the pressure rise in an ATWS transient, the PORVs and SVs are indeed necessary. There are three PORVs and one SV of the pressurizer; because the PORVs work normally in the ATWS transients, they could be lumped into one set in the TRACE model. Besides, the amount of PORVs and SVs of each main steam line are one and five, respectively. The SVs of the main steam line cannot be lumped on account of different set points. The above valves established in the TRACE model based on plant specific parameters, especially the rated flow rates and boundary conditions.

2.6 AMSAC

In order to conform to 10 CFR 50.62, Maanshan NPP has set up AMSAC that is diverse from the reactor trip system to automatically initiate the auxiliary feedwater system and initiate a turbine trip under conditions indicative of an ATWS. AMSAC is a backup system that initiates if the RTS (Reactor Trip System) and ESFAS (Engineered Safety Features Actuation System) fail to work following an ATWS. It will be initiated under three separate conditions: (1) Low steam generator water level, (2) Low main feedwater flow, (3) Main feedwater pump trip or main feedwater valve closure. It has a delay time for making sure that the RPS is failed, the amount depends on the core power at that time which is about 30 seconds at hot full power.

Table 1 The parameters of the pressurizer for Maanshan NPP

Parameter	
Pressurizer volume	39.64 m^3
Full power water level	56.5%
Pressurizer flow length	11.74 m
Spray valve max. flow rate (total 2 valves)	44 L/sec
Heater (proportional heater and backup heater)	1400 W
Safety valve set point	17.24 MPa
Power-operated relief valve set point	16.20 MPa

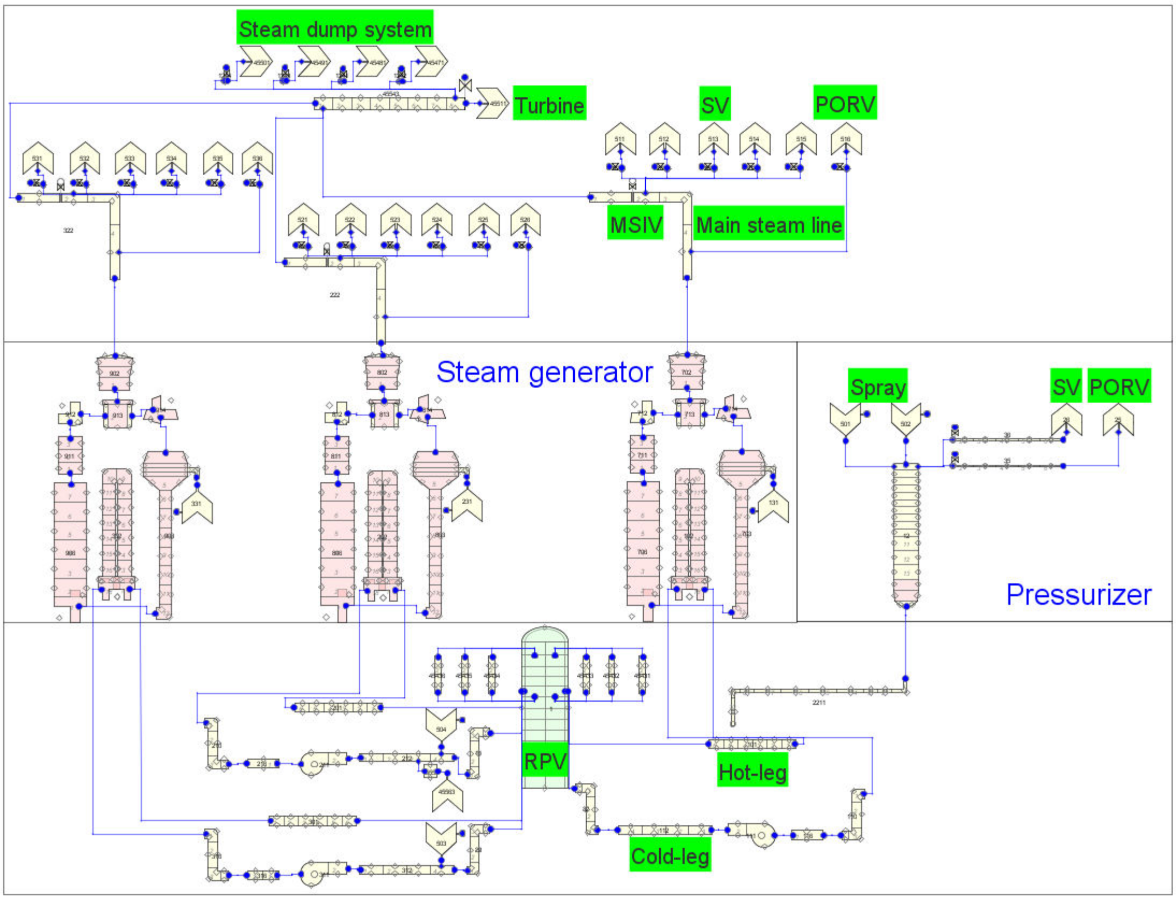

Figure 1　The TRACE model for Maanshan NPP

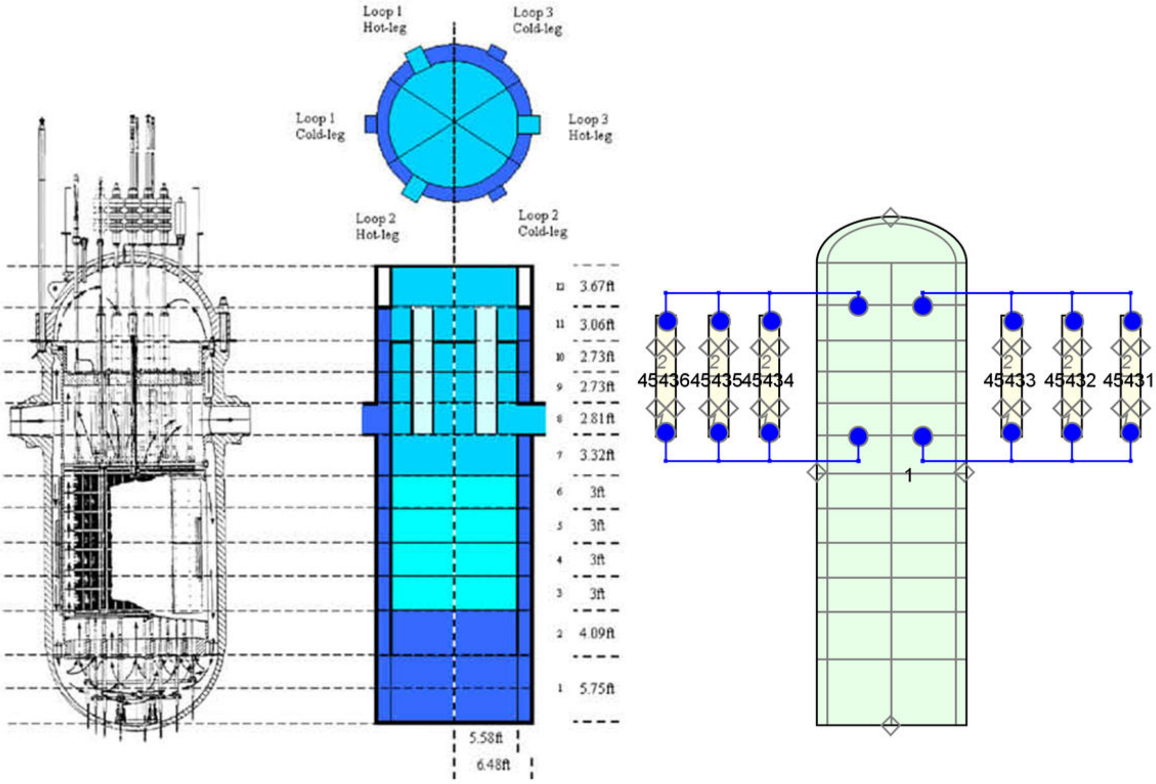

Figure 2 Cross section and the TRACE model of vessel for Maanshan NPP

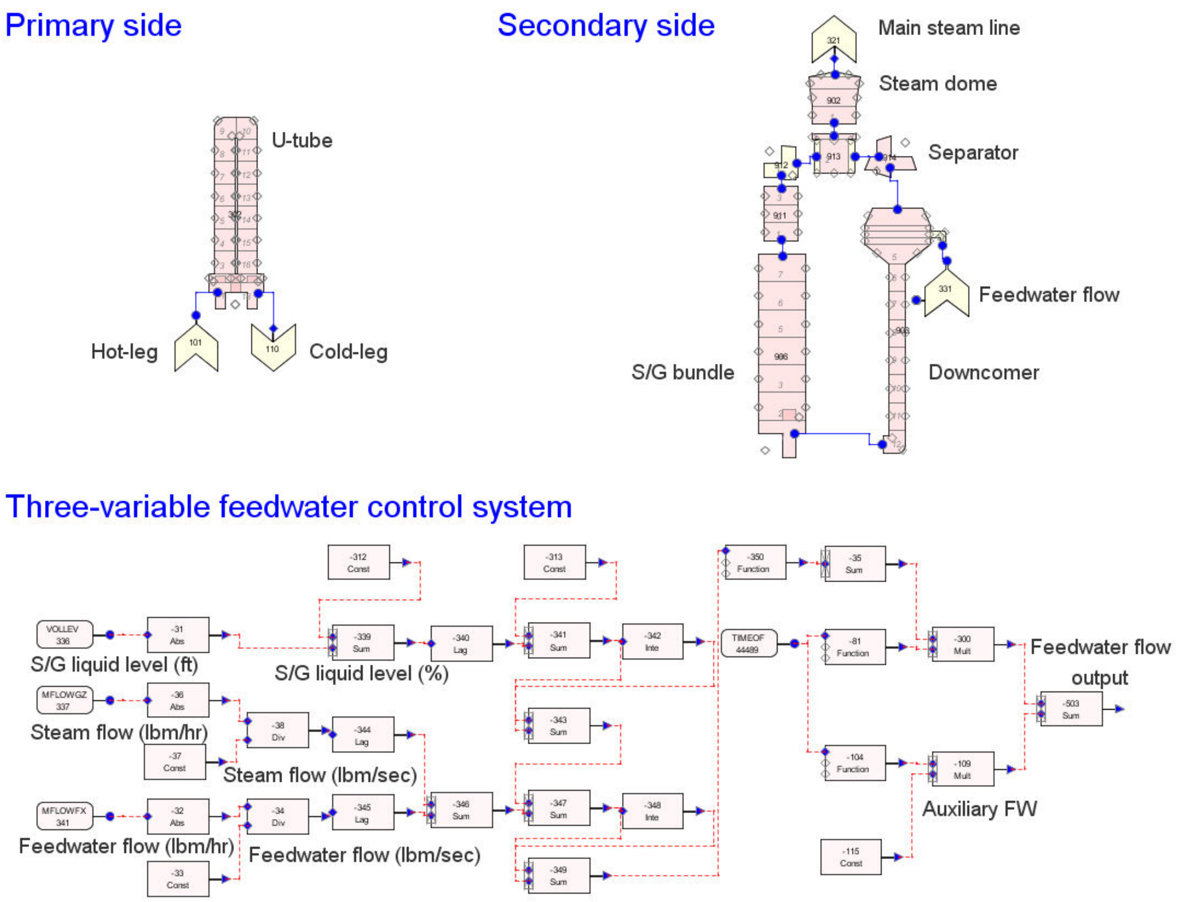

Figure 3　TRACE model of steam generator and feedwater control system for Maanshan NPP

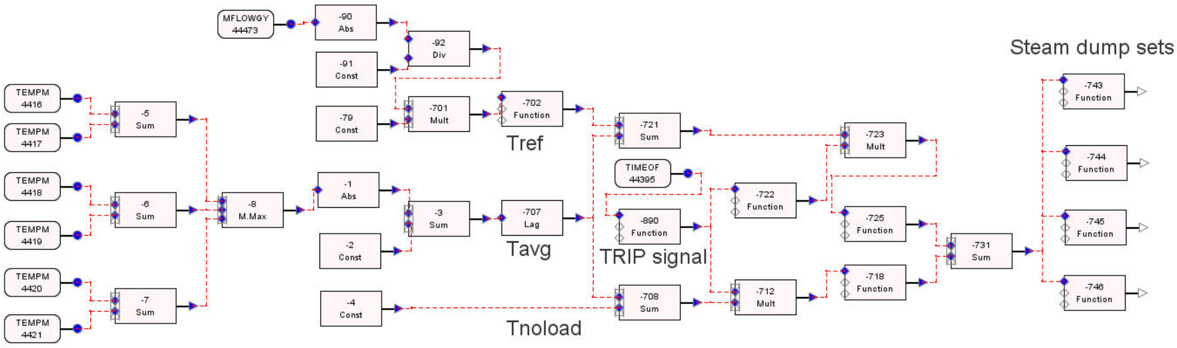

Figure 4　The TRACE model of steam dump control system for Maanshan NPP

3. DESCRIPTION AND ASSUMPTIONS OF LONF ATWS

The PWR sequence starts with an anticipated transient and the electrical or mechanical failure of the RPS. In a PWR, the ATWS transient results in a RCS pressure rise, the magnitude and timing of which is dependent on the MTC, the relief capacity, and the energy removal capacity of the steam generators [9].

Before any transient analysis using TRACE whole plant model, a consistent set of parameters used in TRACE must be obtained from the process of steady-state initialization. The parameters computed from steady-steady initialization such as feedwater/steam flows and water level of the steam generators, water level and pressure of the pressurizer. And the hot-leg temperatures were then compared with real plant data. After completing the steady-state initialization, the ATWS transient analysis began with point-kinetic power calculation which reactivity coefficients defined by the parameters: fuel temperature coefficient (Doppler coefficient), coolant temperature coefficient (optional), gas volume coefficient (optional), and solute mass coefficient (optional).

The followings are the general assumptions of Maanshan ATWS:

(1) LONF transient was initiated at 10 sec after the beginning of calculation
(2) Main feedwater flow descended to zero in the first four seconds of transient
(3) According to the design basis of AMSAC, the maximum time of signal delay at full power was 30 sec; therefore, the turbine was assumed to trip at 40 sec, and the AFW was initiated at 70 sec
(4) Normal operation of pressurizer pressure control, including heaters, spray, PORV and SV
(5) Normal operation of main steam valves, including PORV, SV and steam dump system
(6) No credit for automatic reactor trip
(7) No credit for automatic control rod insertion as reactor coolant temperature rises

Besides, sensitivity studies as well as MTC variations, RCP trip, failure of partial motor-driven AFW were taken into consideration specifically.

3.1 MTC Variations

The MTC taken for calculation of Maanshan NPP in early days were -12.6 pcm/K (-7 pcm/℉) at 1% burn-up and -14.4 pcm/K (-14.4 pcm/℉) at 10% burn-up. Since 1993, the Taiwan Power Company (TPC) in order to extend fuel cycle to 18 months, the maximum MTC was tightened to -7.2 pcm/K (-4 pcm/℉). Therefore, we took these MTC settings into account to assess the resulting pressure, and chose maximum part as the conservative condition of ATWS calculations.

3.2 RCP Trip

RCP trip may result in lower RCS pressure and coast down of loop flow that causes lower capacity of heat removal from reactor core; furthermore, Lower RCS pressure also brings about lower saturation temperature. The set-point of RCP trip was set to be 3.3 K (6 ℉) of inlet subcooling to prevent impeller cavitation, and the pumping curve of RCP after trip was calculated by TRACE built-in Westinghouse Curves.

3.3 Failure of Partial Motor-Driven AFW

The AFW system of Maanshan NPP has two motor-driven pumps and one turbine-driven pump and relative piping to each steam generators, the water sources can be drained from either condensate storage tank (CST), demineralized water storage tank (DST) or raw water (Figure 5). AFW played the important role to maintain inventory of steam generators to fulfill the capacity of heat removal from primary side. Hence, one motor-driven AFW was cancelled due to electrical or mechanical failure as valve closure to simulate partial loss of AFW.

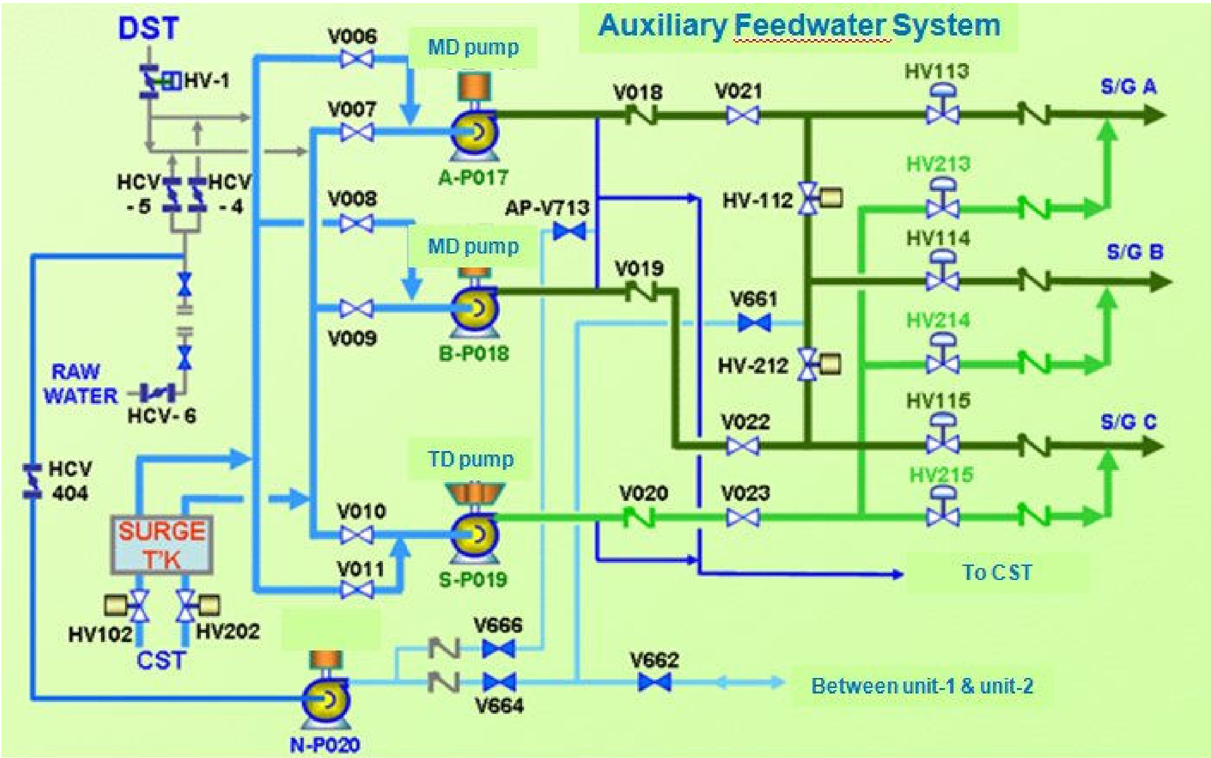

Figure 5 AFW system of Maanshan NPP[10]

4. RESULTS AND DISCUSSIONS

4.1 MTC Variations

Table 2 shows the sequences of events, Transients began at 10 sec with following loss of feedwater flow in 4 seconds that caused the steam generator inventory to decrease gradually (Figure 6). Main feedwater pump trip brought about AMSAC standby. The secondary side began to lose its heat-sink property because of decreasing heat removal from steam generators and made temperature at primary side rise (Figure 7 and Figure 8). Turbine tripped at 40 sec that initiated by AMSAC, loss of ultimate heat sink induced system pressure rise rapidly. The pressurizer spray system began to drain coolant from cold-leg to the upper plenum of pressurizer to mitigate the rise of temperature and pressure at 41 sec. Furthermore, the RCS temperatures rose to their peak values that led coolant density in the primary side descend. Density drop of primary coolant made its volume increase to fill all the room of pressurizer (Figure 9); meanwhile, the pressurizer valves as well as main steam line valves were initiated to mitigate the pressure rise. AFW entered steam generator at 70 sec to supply inventory, but the RCS pressures still rose to their peak values of the ATWS transients (Figure 10, 20.92 MPa for -7.2 pcm/K, 19.33 MPa for -12.6 pcm/K, and 18.99 MPa for -14.4 pcm/K). Although the timing for pressurizer with water filled was later with more negative MTC, the peak RCS pressure reached earlier. Because more negative MTC causes slower pressure rise, which means slower growth of pressurizer water level and stronger mitigation of pressure rise. After 300 sec, there were several small fluctuations for -12.6 pcm/K and -14.4 pcm/K because of the opens of pressurizer PORVs (set-point at 16.20 MPa). With the effort of PORVs, SVs, pressurizer spray system, and descend of core power (Figure 11, the core power decreased to about 15% of hot full power in 5 min), the RCS pressures were kept within 22.06 MPa. For the following discussions, we took -7.2 pcm/K as the conservative condition of MTC.

Table 2 Sequences of events of LONF ATWS – MTC variations

	MTC setting (pcm/K)		
Transient (sec)	**-7.2**	**-12.6**	**-14.4**
Transient initiates	10		
Main feedwater trips	10 – 14		
Turbine trips	40		
Pressurizer sprays actuate	41	41	41
Pressurizer PORVs open	43	44	44
Main steam line PORVs open	44	44	44
Main steam line SVs open	52	51	51
Full AFW flow actuates	70		
Pressurizer safety valves open	79	81	82
Pressurizer fills with water	83	85	86
Peak RCS pressure reaches	112	109	109

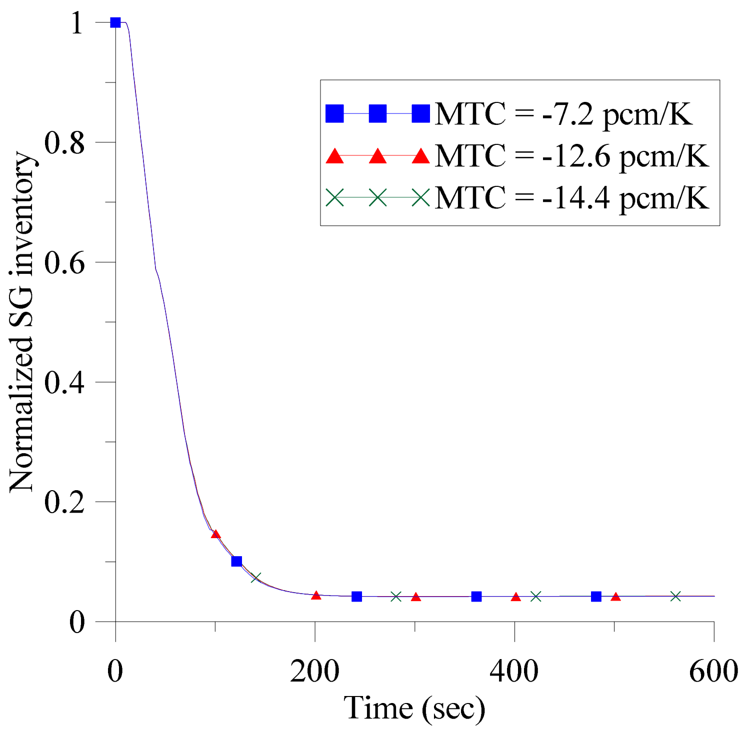

Figure 6 MTC variations – steam generator inventory

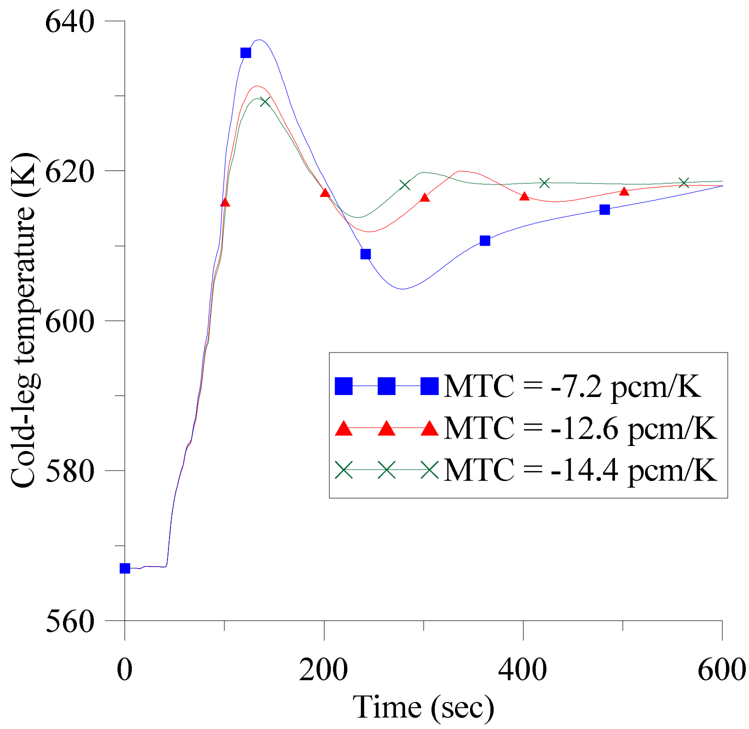

Figure 7 MTC variations – cold-leg temperature

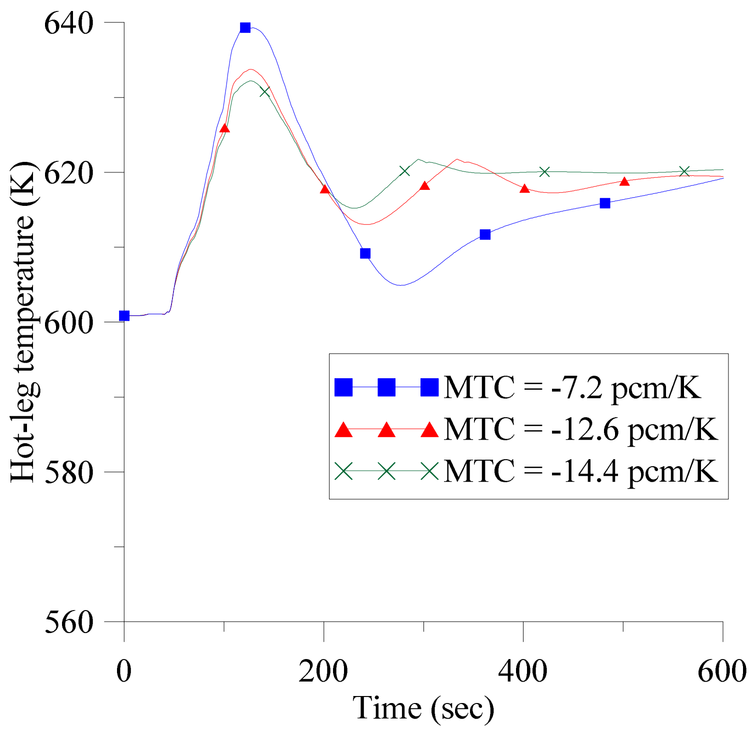

Figure 8 MTC variations – hot-leg temperature

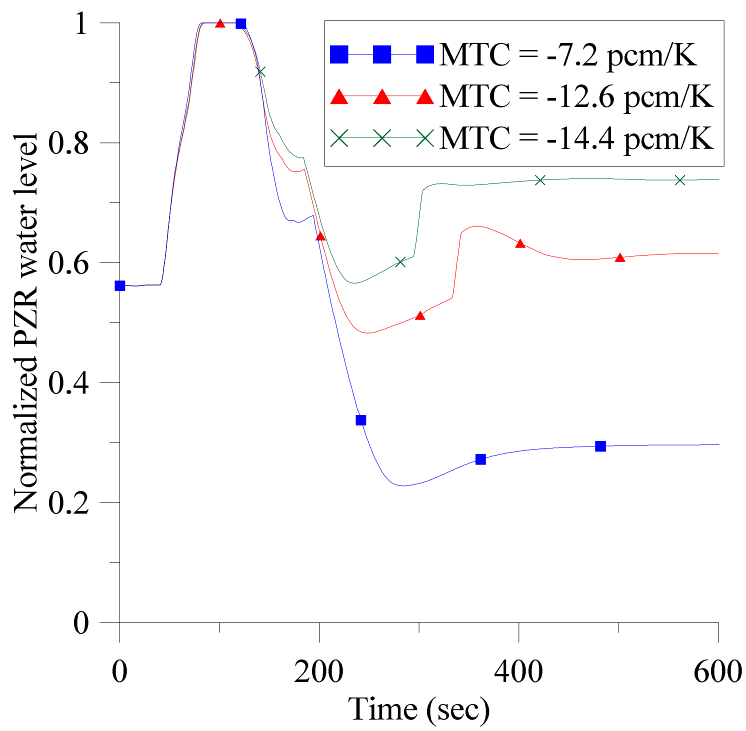

Figure 9 MTC variations – pressurizer water level

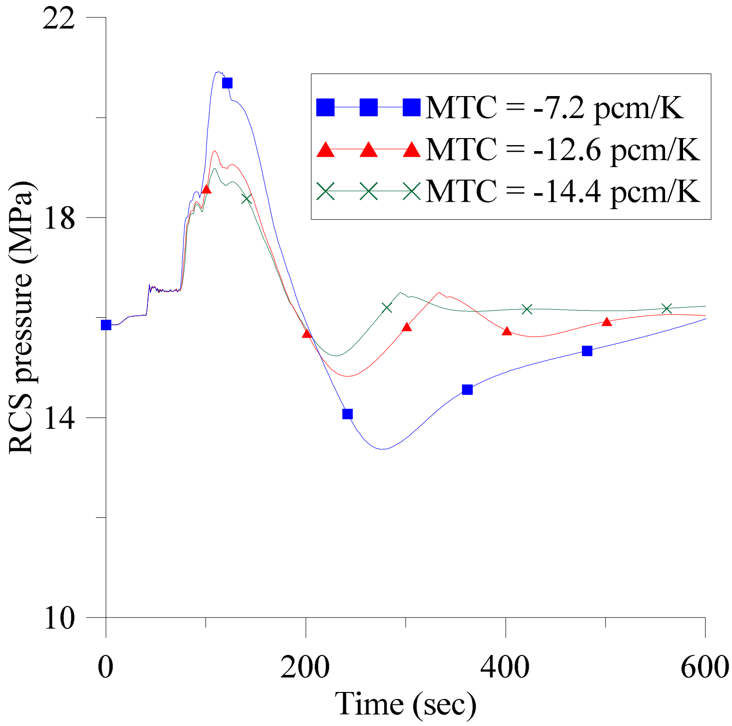

Figure 10 MTC variations – RCS pressure

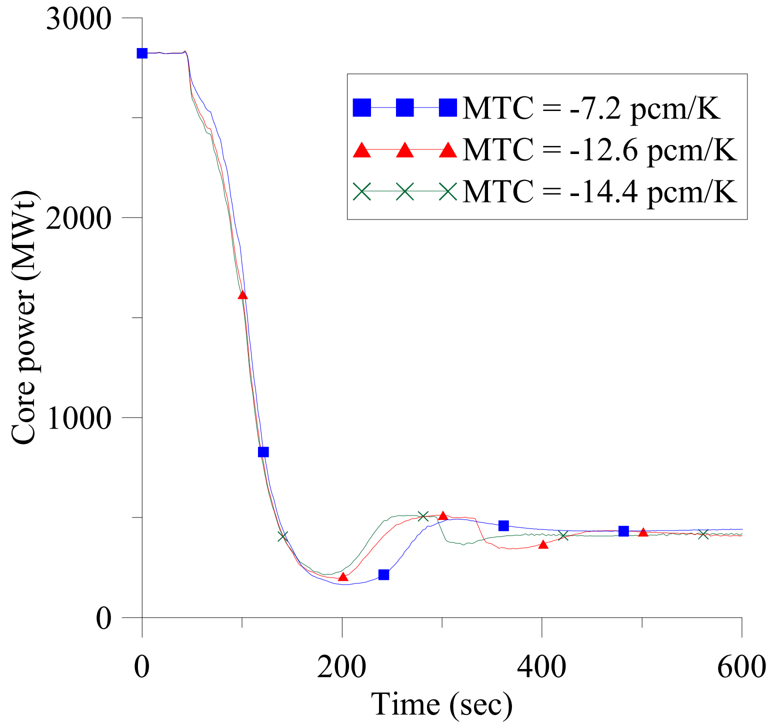

Figure 11 MTC variations – core power

4.2 RCP Trip

Table 3 shows the sequences of events, Transient began at 10 sec with following loss of feedwater flow in 4 seconds that caused the steam generator inventory to decrease gradually (Figure 12). Main feedwater pump trip brought about AMSAC standby. The secondary side began to lose its heat-sink property because of decreasing heat removal from steam generators and made temperature at primary side rise (Figure 13 and Figure 14). Turbine tripped at 40 sec that initiated by AMSAC, loss of ultimate heat sink induced system pressure rise rapidly. The pressurizer spray system began to drain coolant from cold-leg to the upper plenum of pressurizer to mitigate the rise of temperature and pressure at 41 sec. Furthermore, the RCS temperatures rose to their peak values that led coolant density in the primary side descend. Density drop of primary coolant made its volume increase to fill all the room of pressurizer (Figure 15); meanwhile, the pressurizer valves as well as main steam line valves were initiated to mitigate the pressure rise. AFW entered steam generator at 70 sec to supply inventory, the RCS pressures rose to its peak value of 20.92 MPa (Figure 16) and the core power descended to about 18% (Figure 17) of hot full power in 5 min. As a result of timing of RCP trip was behind that of peak pressure (Figure 18), RCP trip had done no effort to pressure rise that resulted in the same value as no RCP trip. But RCP trip made RCS temperature descend which resulted in less mass dumped by steam dump system (Figure 19), the accumulated heat evaporated coolant that brought about lower core water level to approach TAF (Top of Active Fuel) that threatened the integral of fuel rods (Figure 20).

Table 3 Sequences of events of LONF ATWS – RCP trip

Transient (sec)	No RCP trip	RCP trip
Transient initiates	10	
Main feedwater trips	10 – 14	
Turbine trips	40	
Pressurizer sprays actuate	41	41
Pressurizer PORVs open	43	43
Main steam line PORVs open	44	44
Main steam line SVs open	52	52
Full AFW flow actuates	70	
Pressurizer safety valves open	79	79
Pressurizer fills with water	83	83
Peak RCS pressure reaches	112	112
RCPs trip	None	124

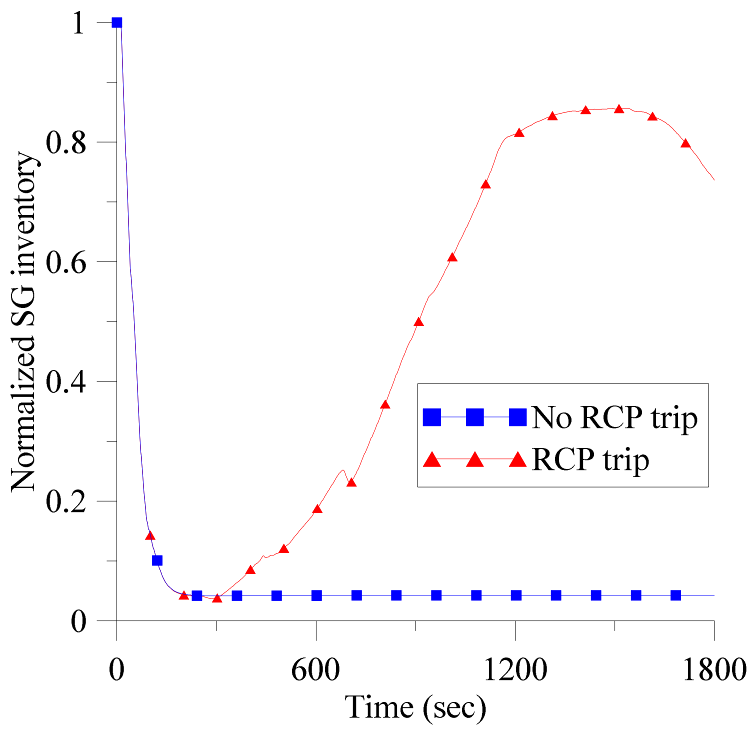

Figure 12 RCP trip – steam generator inventory

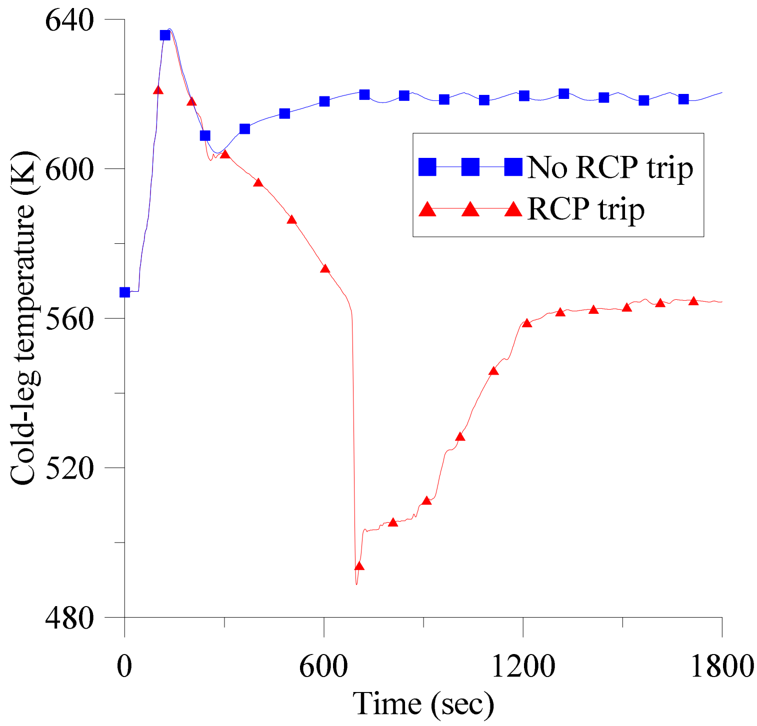

Figure 13 RCP trip – cold-leg temperature

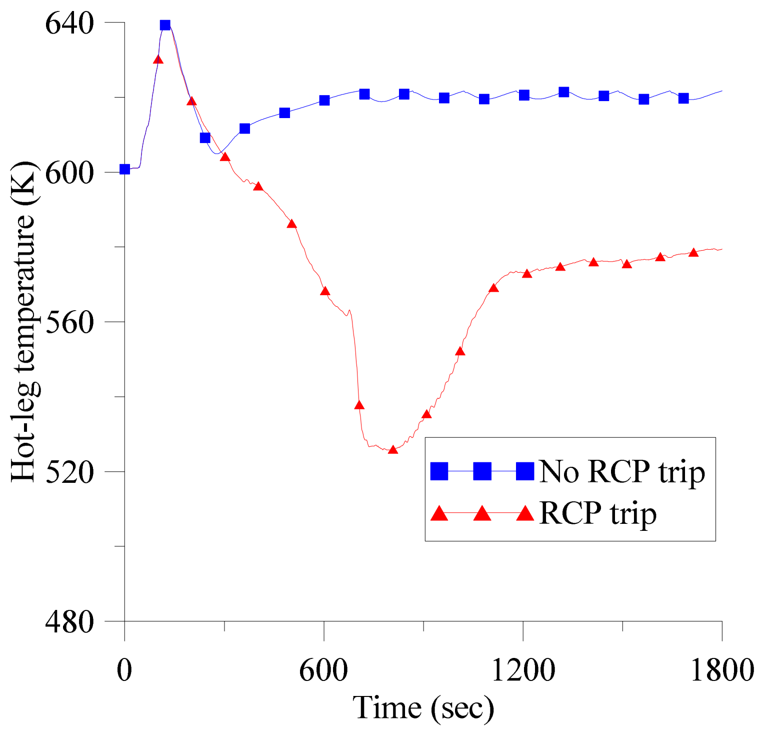

Figure 14 RCP trip – hot-leg temperature

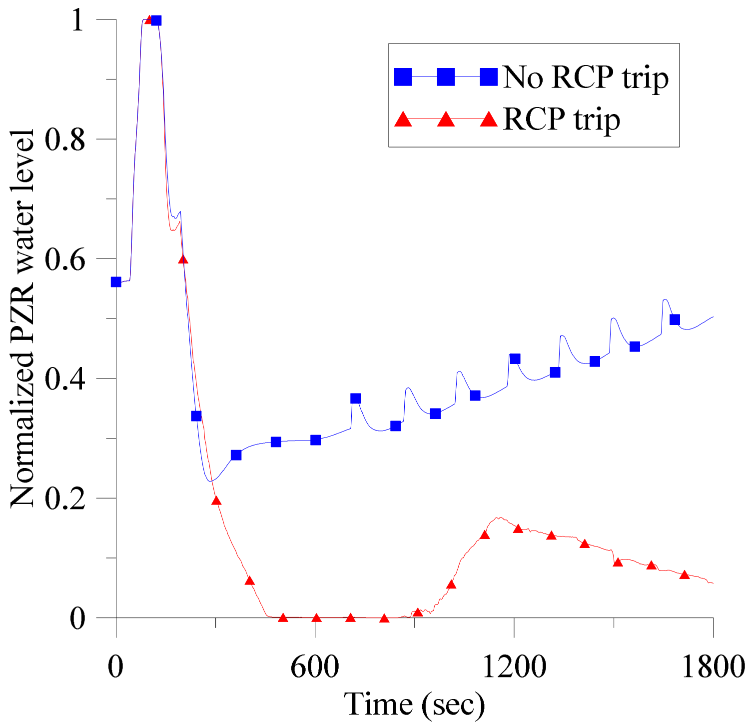

Figure 15 RCP trip – pressurizer water level

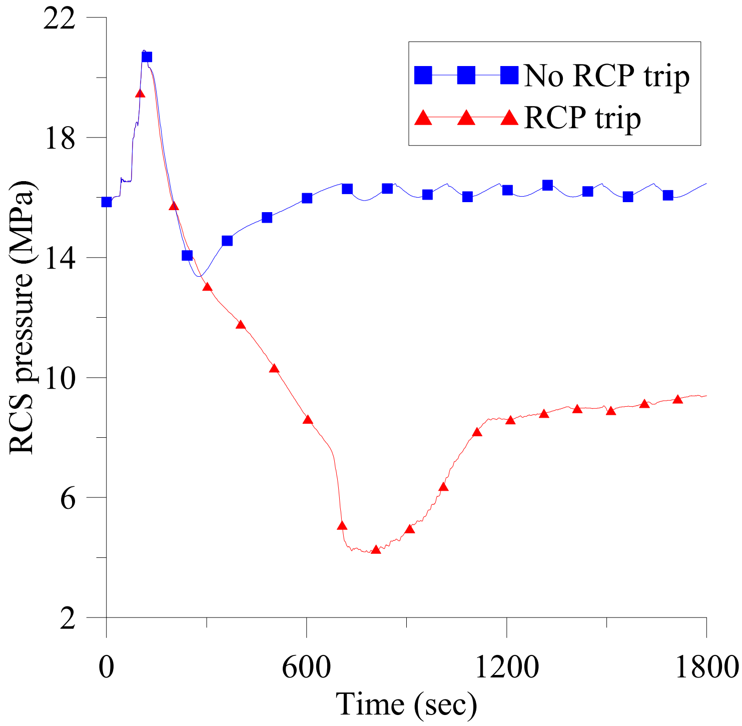

Figure 16 RCP trip – RCS pressure

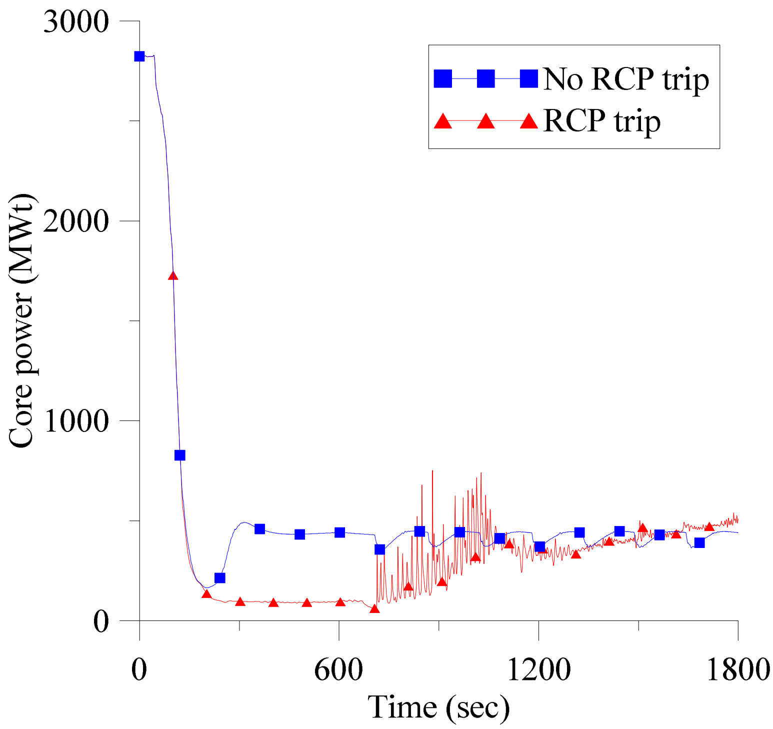

Figure 17 RCP trip – core power

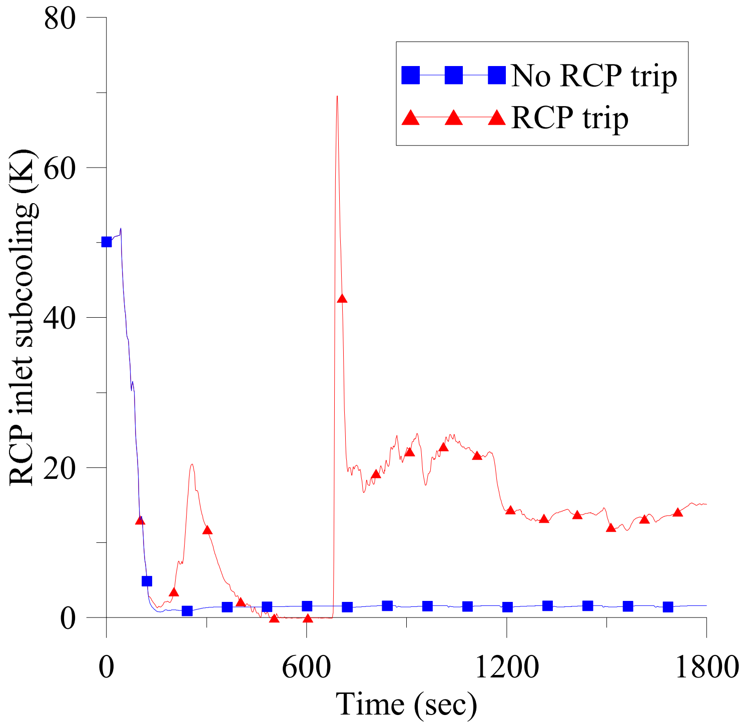

Figure 18 RCP trip – RCP inlet subcooling

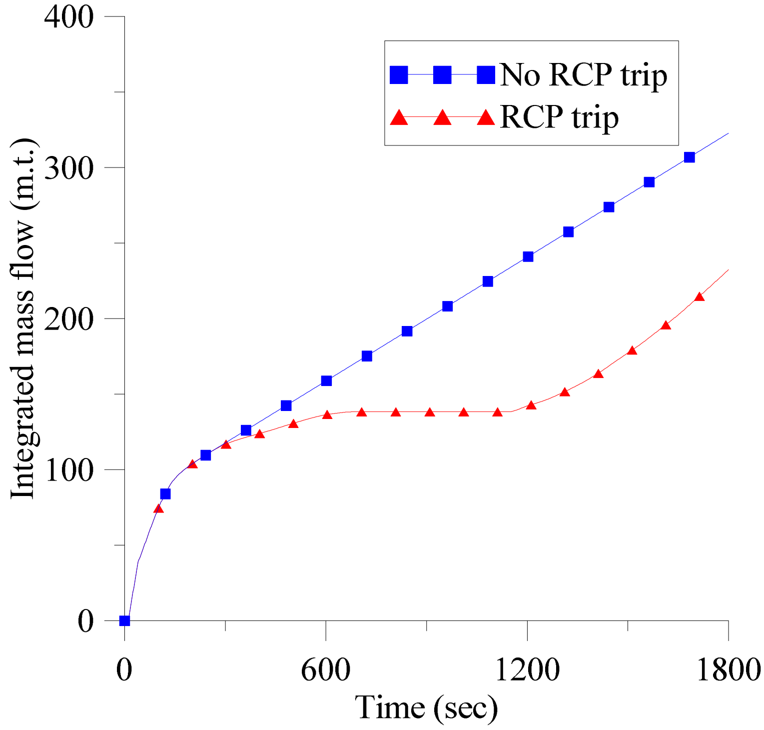

Figure 19 RCP trip – integrated mass flow of steam dump system

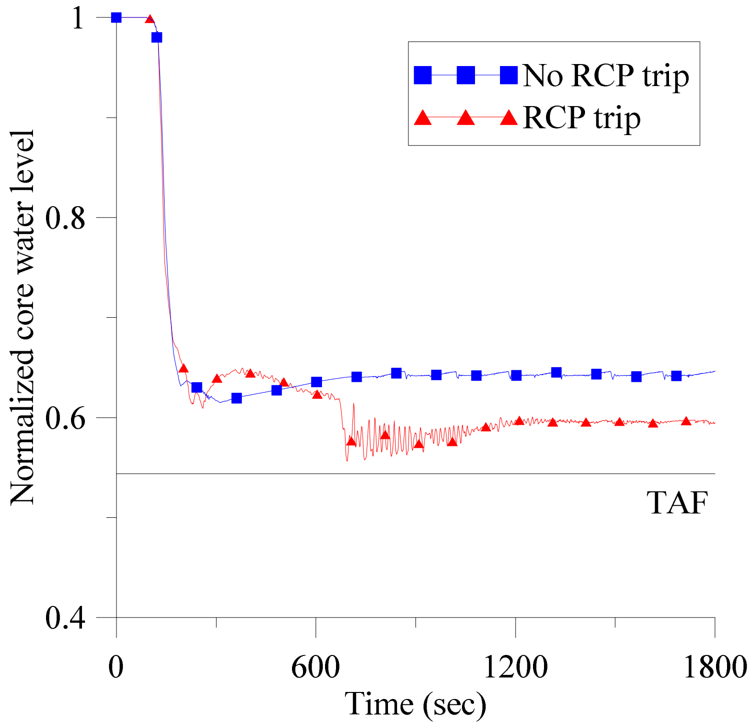

Figure 20 RCP trip – core water level

4.3 Failure of Partial Motor-Driven AFW

Table 4 shows the sequences of events, Transient began at 10 sec with following loss of feedwater flow in 4 seconds that caused the steam generator inventory to decrease gradually (Figure 21). Main feedwater pump trip brought about AMSAC standby. The secondary side began to lose its heat-sink property because of decreasing heat removal from steam generators and made temperature at primary side rise (Figure 22 and Figure 23). Turbine tripped at 40 sec that initiated by AMSAC, loss of ultimate heat sink induced system pressure rise rapidly. The pressurizer spray system began to drain coolant from cold-leg to the upper plenum of pressurizer to mitigate the rise of temperature and pressure at 41 sec. Furthermore, the RCS temperatures rose to their peak values that led coolant density in the primary side descend. Density drop of primary coolant made its volume increase to fill all the room of pressurizer (Figure 24); meanwhile, the pressurizer valves as well as main steam line valves were initiated to mitigate the pressure rise. AFW generated by one turbine-driven pump and one motor-driven pump entered steam generator at 70 sec to supply inventory, the RCS pressures rose to its peak value of 21.80 MPa (Figure 25) and the core power descended to about 10% (Figure 26) of hot full power in 5 min. Trip of one motor-driven AFW pump caused decrease of AFW flow that resulted in less heat removal capacity by steam generator and further rise of RCS temperature and pressure. During the opening of pressurizer PORVs and SVs, the increase of pressure difference made larger amount of coolant dump through valves (Figure 27). Therefore, the pressurizer water level (Figure 24) and core water level (Figure 28) became lower as coolant density shrunk behind peak pressure. Furthermore, the animation of the TRACE model is presented using the animation function of SNAP/TRACE interface with the analysis results. The animation model of Maanshan NPP is shown in Figure 29.

Table 4 Sequences of events of LONF ATWS – Failure of partial MDAFW

Transient (sec)	No AFW trip	One MDAFW trip
Transient initiates	10	
Main feedwater trips	10 – 14	
Turbine trips	40	
Pressurizer sprays actuate	41	41
Pressurizer PORVs open	43	43
Main steam line PORVs open	44	44
Main steam line SVs open	52	51
Full AFW flow actuates	70	
Pressurizer safety valves open	79	79
Pressurizer fills with water	83	83
Peak RCS pressure reaches	112	112

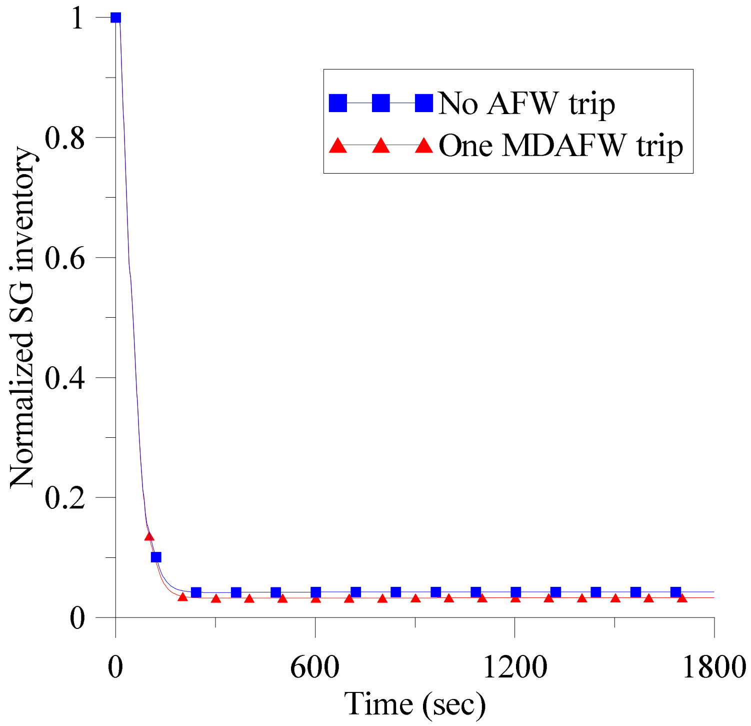

Figure 21 Failure of partial MDAFW – steam generator inventory

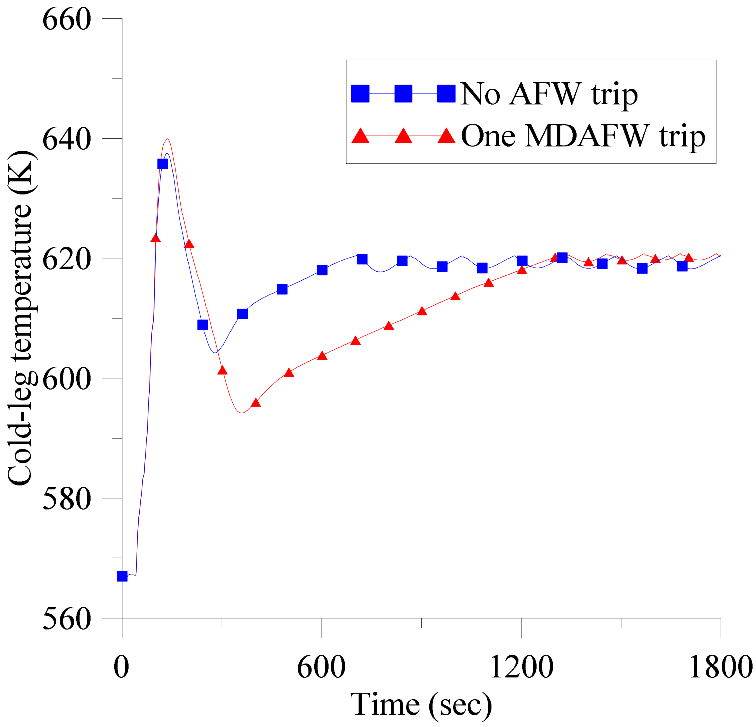

Figure 22 Failure of partial MDAFW – cold-leg temperature

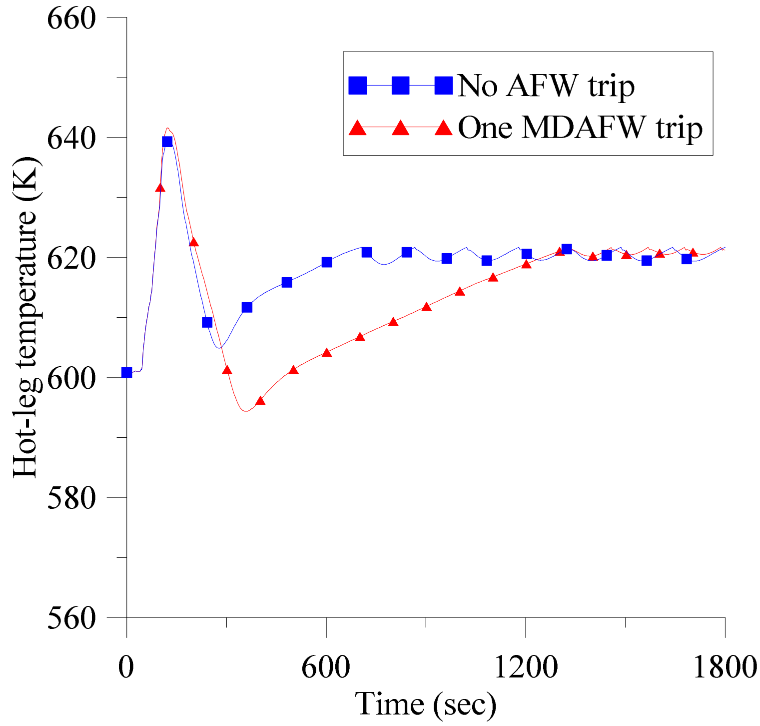

Figure 23 Failure of partial MDAFW – hot-leg temperature

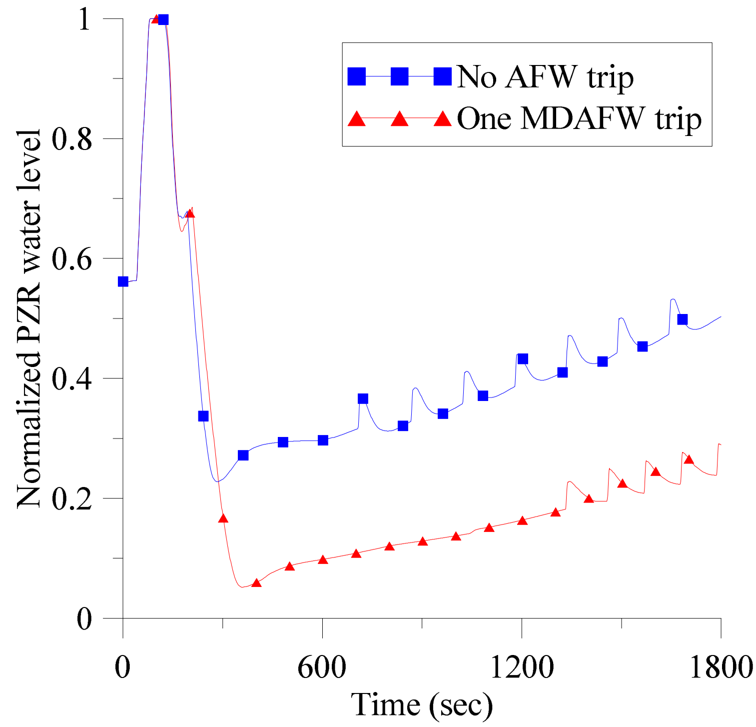

Figure 24 Failure of partial MDAFW – pressurizer water level

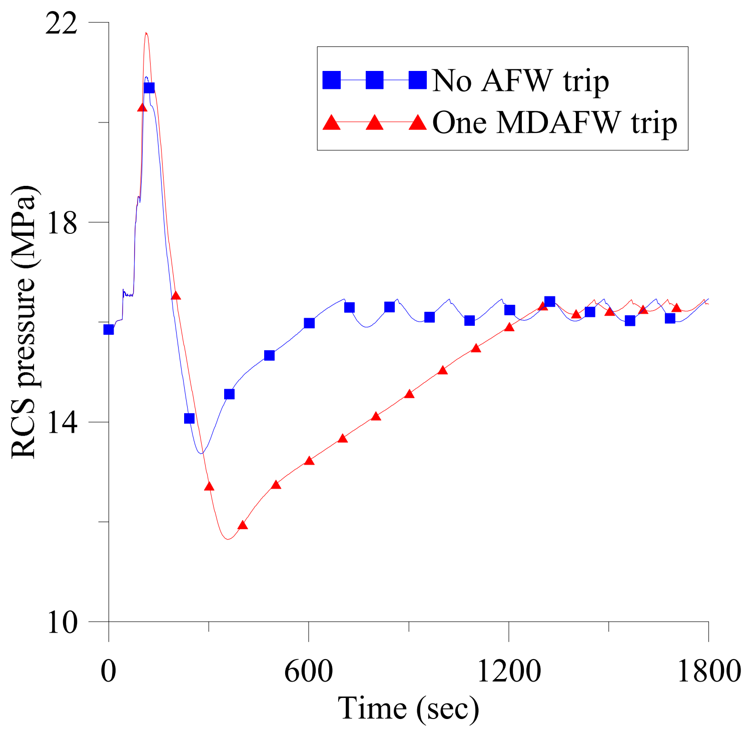

Figure 25   Failure of partial MDAFW – RCS pressure

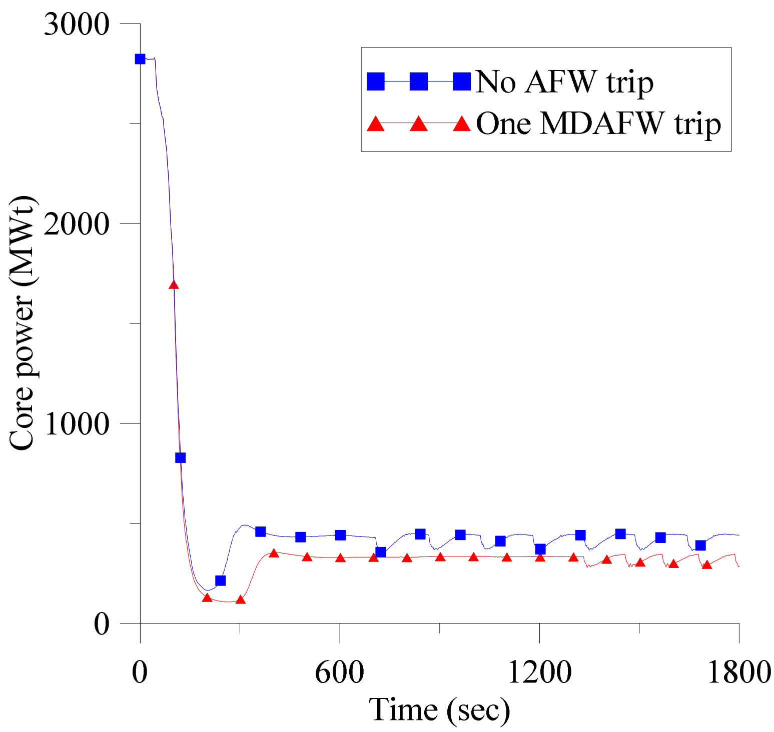

Figure 26   Failure of partial MDAFW – core power

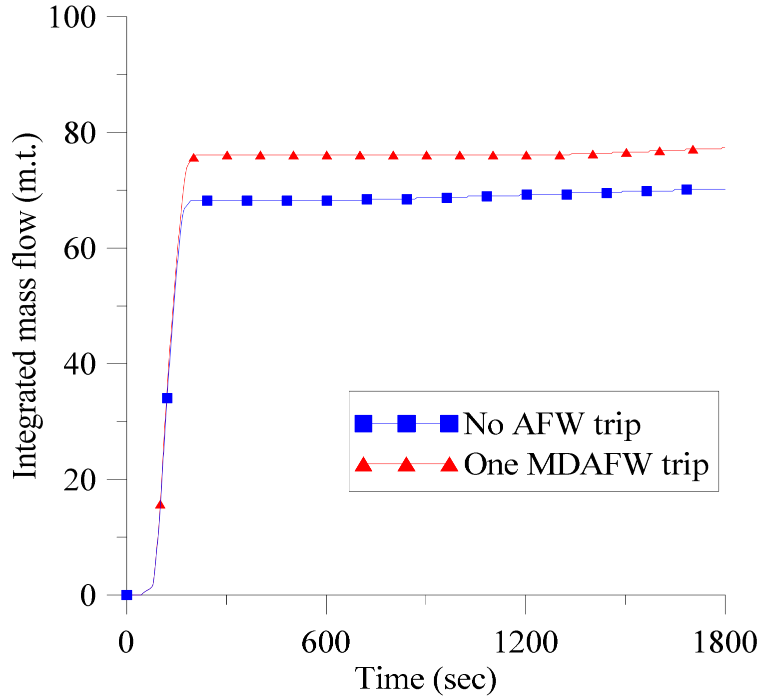

Figure 27 Failure of partial MDAFW – integrated mass flow of pressurizer PORVs and SVs

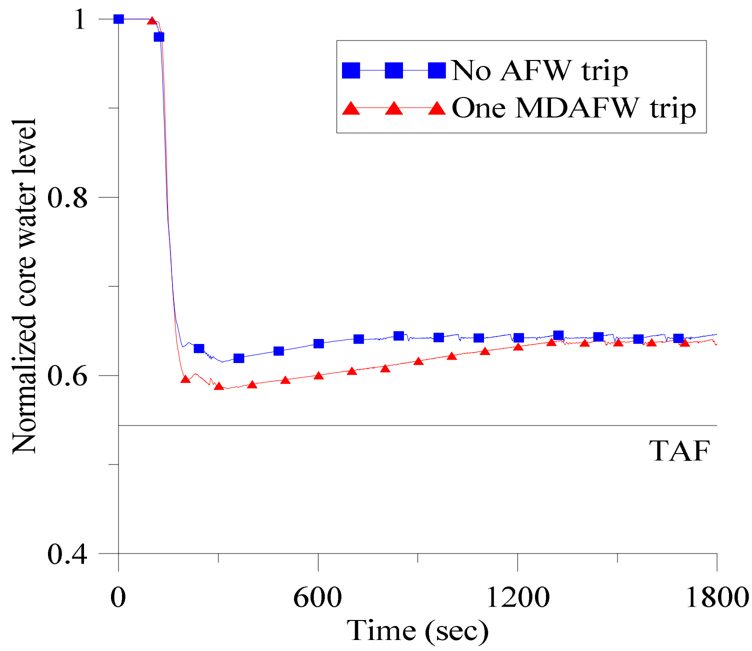

Figure 28 Failure of partial MDAFW – core water level

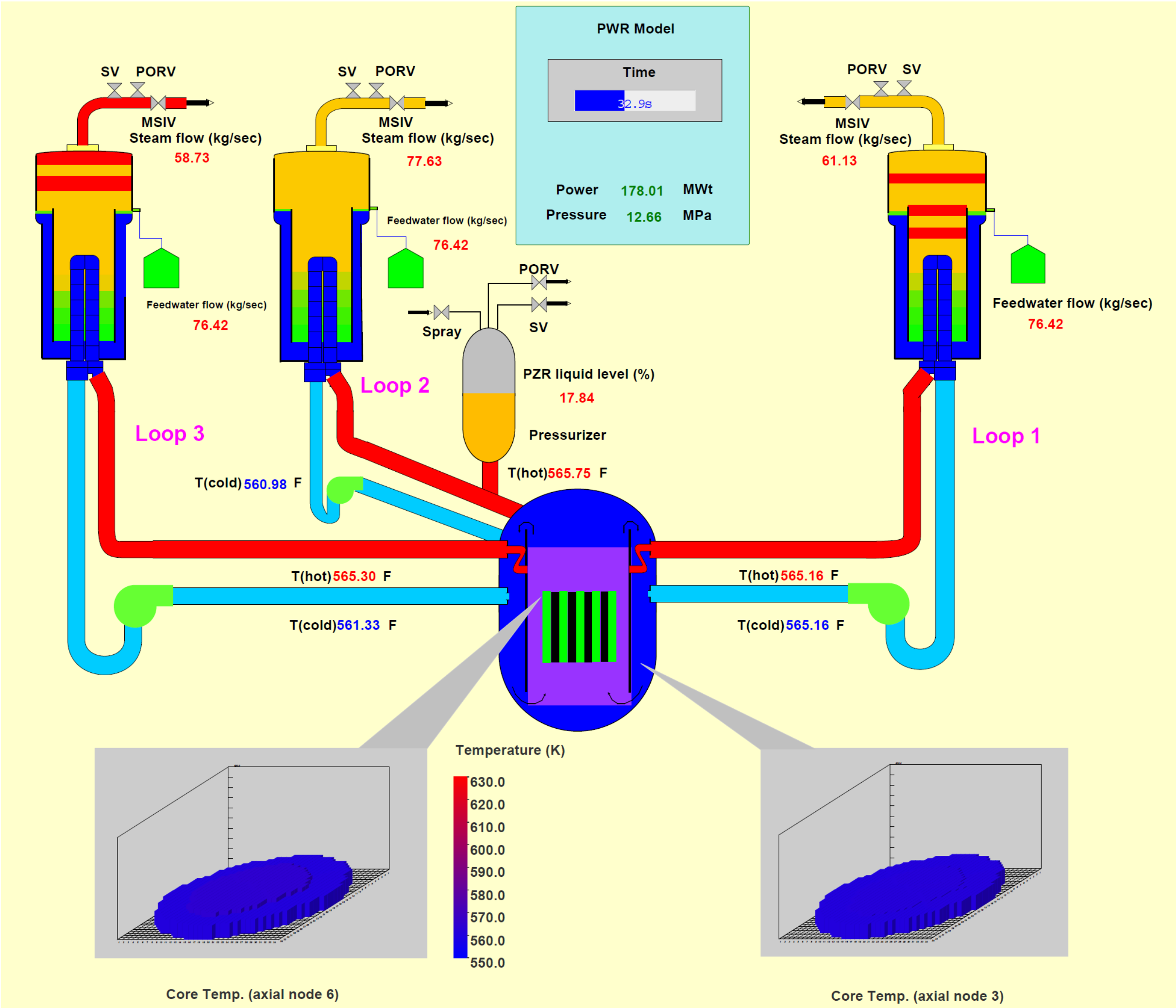

Figure 29　The animation model of Maanshan NPP

5. CONCLUSIONS

By using SNAP/TRACE, this study predicts the transient phenomena of LONF ATWS and assesses the peak pressures. According the simulation results, the peak pressures were 20.92 MPa for -7.2 pcm/K, 19.33 MPa for -12.6 pcm/K, and 18.99 MPa for -14.4 pcm/K; moreover, even the conservative MTC condition of -7.2 pcm/K was employed, the RCS pressure could still keep within the ASME Code Level C service limit criteria of 22.06 MPa. The negative MTC, normal operations of PORVs and SVs, and heat removal capacity by steam generator (which inventory maintained by AMSAC function and relative facilities) are the important parts to mitigate pressure fluctuations and make coolant cover fuel rods.

In order to simulate further severe situations, we choose RCP trip for primary loop and failure of partial MDAFW for secondary loop that both result in less heat removal during transients. As RCPs trip at 3.3 K of inlet subcooling to prevent impeller cavitation, it results in lower RCS pressure (but no effort to peak pressure), but RCP trip brings about lower core water level to approach TAF that threatens the integral of fuel rods. RCPs should be mandatorily kept working as ATWS takes place. AFW is necessary to steam generator inventory, trip of one MDAFW results in less inventory and following pressure rise due to accumulated heat at primary side. It also causes lower coverage of core water level. Therefore, regularly inspecting the facilities of AFW is helpful to mitigation of ATWS.

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10. SUPPLEMENTARY NOTES
K. Tien, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The objective of this study is to utilize TRACE code to analyze the reactor coolant system (RCS) pressure transients under LONF (Loss of Normal Feedwater) ATWS (Anticipated Transient Without Scram) for Maanshan PWR. TRACE is an advanced thermal hydraulic code for nuclear power plant safety analysis which is developed by U.S. NRC. Maanshan nuclear power plant (NPP) is a Westinghouse three-loop PWR in Taiwan. The rated core thermal power of Maanshan with MUR-PU (Measurement Uncertainty Recapture Power Uprate) is 2822 MWt. According to Westinghouse anticipated transients without trip report, LONF ATWS was regarded as the most severe plant condition. The ASME Code Level C service limit criteria of 22.06 MPa (3200 psig) was assumed to be an unacceptable plant condition in SECY-83-293.
In order to conform to 10 CFR 50.62, Maanshan NPP has set up AMSAC that is diverse from reactor trip system to automatically initiate the auxiliary feedwater system and initiate a turbine trip under conditions indicative of an ATWS. Since the ATWS analysis is not specified in the FSAR, we use TRACE code to assess the RCS pressure for Maanshan NPP. The results indicate that RCS pressure could keep within 22.06 MPa with sufficient negative moderator temperature coefficient (MTC) and normal work of AMSAC and valves.

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