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TRACE Simulation of SBO Accident and Mitigation Strategy in Maanshan PWR

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ABSTRACT

Maanshan Nuclear Power Station is a two-unit Westinghouse three-loop PWR power station. This research studies the simulation of Maanshan SBO accident happened on 18th March, 2001, and thermal-hydraulic phenomena of the plant during station blackout with and without mitigation strategies. The modeling and simulation works were done by using TRACE code, which is a best-estimate thermal-hydraulic system code developed by US NRC. The purpose of using the mitigation strategy during SBO is to cool down NSSS as soon as possible, to keep the fuel covered by water, and not to let peak cladding temperature (PCT) higher than 1088K (1500°F), which is the temperature that metal-water reaction can self-sustain. Actions that considered such as operation of auxiliary feedwater system, depressurization of steam generators (SG), and line-up the alternate water sources such as sea water when regular systems aren't available.

The simulations of mitigation strategies start from normal operation at 100% power then an earthquake is assumed to happen, tsunami strike the site 20 minutes later, and failure of turbine driven auxiliary feedwater is assumed in base cases. Two different basic mitigation strategies were simulated, include (1) SG controlled-depressurization at the time that SBO happen and SG alternate injection after 1 hour of SBO, (2) SG depressurization at the time after 1 hour of SBO that injection is ready. In addition, alternate injection preparation time is further extended to find the longest acceptable value. Reactor coolant pump shaft seal leakage is also considered in all cases.

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 model of Maanshan Nuclear Power Station has been built. In this report, we focus on the TRACE analysis of Maanshan SBO accident.

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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 model of Maanshan Nuclear Power Station is developed by INER.

According to the TRACE user's manual, it 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. Therefore, in the future, NRC has ensured that TRACE will be the main code used in thermal hydraulic safety analysis, without further development of other thermal hydraulic codes such as RELAP5 and TRAC. Besides, the 3-D geometry model of reactor vessel is one of the features of TRACE. It can support a more accurate and detailed safety analysis of NPPs.

Maanshan Nuclear Power Station is a two-unit Westinghouse three-loop PWR power station. This research first analyzes the SBO accident happened on 18 March, 2001 by using TRACE code and compares the results with plant data. The results show good agreement with plant data, and then this input model is used to analyze the mitigation strategies.

The simulations of mitigation strategies start from normal operation at 100% power then an earthquake is assumed to happen, tsunami strike the site 20 minutes later, and failure of turbine driven auxiliary feedwater is assumed in base cases. Two different basic mitigation strategies were simulated, include (1) SG controlled-depressurization at the time that SBO happen with 1 hour SG alternate injection preparation time, (2) SG alternate injection ready in 1 hour, SG depressurization at the time that injection is ready. In addition, alternate injection preparation time is further extended to find the longest acceptable value. Reactor coolant pump shaft seal leakage is also considered in all cases.

The simulation results in this research show that performing steam generator controlled-depressurization at the early stage of accident and, if no regular coolant injection system available, line up the alternate injection system in 3.5 hours after SBO can keep the fuels covered with water therefore maintain the plant in safety condition. In addition, after plant is under controlled at the early stage of accident, onsite operators should recover AC power as quickly as possible so that ECCS can make up RCS inventory loss through RCP seal, and residual heat removal system (RHR) can remove the decay heat in RCS continuously until reactor is at cold shut down.

ABBREVIATIONS

AC	Alternating current
ACC	Accumulator
AFWS	Auxiliary feedwater system
CST	Condensate storage tank
ECCS	Emergency core cooling system
FW	Feedwater
MDAFW	Motor driven auxiliary feedwater
NPS	Nuclear power station
NSSS	Nuclear steam supply system
PCT	Peak cladding temperature
PORV	Power operated relieve valve
PWR	Pressurized water reactor
RCP	Reactor coolant pump
RCS	Reactor coolant system
RHR	Residual heat removal system
SBO	Station blackout
SG	Steam generator
SNAP	Symbolic Nuclear Analysis Package
TAF	Top of active fuel
TDAFW	Turbine driven auxiliary feedwater
TRACE	TRAC/RELAP Advanced Computational Engine
US NRC	United State Nuclear Regulatory Commission

1. INTRODUCTION

Tragedy happened in Fukushima Daiichi, Japan shows significant consequences of the plant facing beyond design basis accident without proper emergency equipment and mitigation strategies. Taiwan is located at the intersection of two tectonic plates where earthquake frequently happen, with the fact that all nuclear power stations sit near the shore, enhancing the capability of dealing with earthquake induced tsunami or other beyond basis accident is a must.

Maanshan Nuclear Power Station is a two-unit Westinghouse three-loop PWR power station operated by Taiwan Power Company since 1984. If an intense earthquake and tsunami hit the plant, the sea water pumps, switch yard, onsite electric systems, emergency diesel generator or its fuel supply may be damaged and hard to recover. Since no AC power available, only turbine driven auxiliary feedwater system (TDAFW) can deliver cold water to steam generators (SG) to maintain the water level inside SG. If TDAFW trip for some reason, water in the SG will boil off eventually, primary side will lose the heat sink. ECCS cannot operate without AC power so that there is no cooling water injection capability in the reactor coolant system (RCS) except passive accumulators (ACC) actived when RCS pressure is lower than ACC nitrogen gas pressure. Under such circumstance without using any mitigation equipment or strategies, core damage will happen within a few hours.

Taiwan Power Company has enhanced the capability of coping with extended station blackout situation by using mitigation strategies and alternate injection systems. In addition to regular ECCS and auxiliary feedwater system, some alternate injection systems such as diesel engine auxiliary feed pump and fire engine pump can also inject water into SG or RCS, but the operating pressure of the alternate systems is much lower than regular system, and onsite operators have to line-up the injection piping manually. The water sources of alternate injection systems can be either CST, raw water reservoir, or sea water. The mitigation strategies for PWR plant that suggested by Taiwan Power Company put emphasis on removing the decay heat rapidly by controlling the steam generator pressure at a lower level while maintaining the steam generator water level at the same time by using any kind of injection method. If decay heat can be removed successfully via steam generators, RCS pressure will not build up to the opening set point of power operated relieve valves (PORV) that installed on the pressurizer so that no loss of inventory inside RCS. The purpose of using this kind of mitigation strategies is to bring the plant to safe condition as soon as possible and to keep the fuel covered with water.

This research first analyzes the SBO accident happened on 18 March, 2001 by using TRACE code and compares the results with plant data. The results show good agreement with plant data, and then this input model is used to analyze the mitigation strategies. Two basic strategies are simulated in this study, main different between them is the steam generator depressurization strategy. Sensitivity studies on steam generator injection time and the effect of reactor coolant pump (RCP) seal leakage are also performed.

2. VERIFICATION OF MAANSHAN NPS SBO ACCIDENT WITH TRACE CODE

2.1 Introduction to Maanshan NPS SBO Accident

During spring season in Taiwan, salty wind from the ocean can degrade the insulation of power transmission line and causing the instability of off-site power in nearby nuclear power station. On March 17th, 2001, 3:23 am, 345 kV off-site power line was lost due to seasonal salty wind and 161 kV off-site power was remained available. Unit 1 reactor tripped and was maintained at hot standby condition by operators.

At 0:46 am, March 18th, a malfunctioned breaker in on-site AC power electric system accidentally grounded, which produced electric arc that damaging other electric systems. Emergency 4.16 kV bus train A and B were both loss of power supply which is a station blackout situation. At 0:57 am, turbine driven auxiliary feedwater system (TDAFW) started automatically to provide cold water into steam generators.

At 0:58 am, reactor operators started to initiate the emergency operating procedure (EOP) to depressurize the steam generator. Auxiliary feedwater flow rate, steam generator pressure and steam generator water level ware controlled and maintained manually by the operators. At 2:54 am, the emergency diesel generator successfully supplied AC power to emergency 4.16 kV bus B, SBO situation was terminated [1].

Duration of SBO is about 2 hours, starts from 0:46 am to 2:54 am, March 18th, and the temperature and pressure of reactor decreased from 564 K, 15.3 MPa to 472 K, 4.2 MPa respectively. Fuels were covered with water and no radioactive materials were released during the whole accident. A brief accident scenario is shown in Table 1. The verification of SBO accident is done by TRACE code with Maanshan input model, and the simulation starts from 0:30 am to 3:30 am, March 18th. TRACE input model and simulation results are introduced in the following sections. The simulation results are compared against measured data in Maanshan unit 1, and further studies on Maanshan SBO mitigation strategy are done by this input model.

Table 1 Maanshan SBO accident scenario

Time (hr)	Simulation Time (hr)	Event
0	--	345 kV off-site power lost Reactor trip
21.12	0	Simulation start with hot standby condition
21.38	0.26	Breaker failure (SBO)
21.57	0.45	Turbine driven auxiliary feedwater (TDAFW) start
21.58	0.46	Initiate EOP 570.20 (SG & RCS cooling)
23.52	2.4	SBO terminated
24.12	3	End of simulation

2.2 Description of Maanshan NPS TRACE Model

The computer code used in this research is TRACE (TRAC/RELAP Advanced Computational Engine) V5.0p3 which is a best-estimate thermal-hydraulic system code developed by US NRC, and the input model is edited by using SNAP (Symbolic Nuclear Analysis Package) V2.2.1. Maanshan TRACE base model contains 69 hydraulic components, 380 control blocks, 34 heat structures and 2 power components. Main components including one 3-D vessel, three RCS loops, one pressurizer, three steam generators and basic plant control systems such as 3-element feedwater control, pressurizer spray, pressurizer level and heater control, and steam dump control.

The 3-D vessel component contains 2 radial rings, 6 azimuthal sectors and 12 axial levels. The outer radial ring represents downcomer region and the reactor core is placed in the inner radial ring from axial level 3 to axial level 6. Six control rod guide tubes are connected above the core region. Nuclear fuels are modeled by 6 heat structures each represents 6908 average fuel rods that uniformly placed in 6 azimuthal sectors. Each RCS loop contains hot leg piping, steam generator U-tube, crossover piping, reactor coolant pump, cold leg piping, accumulator tank and accumulator check valve. Pressurizer and pressurizer surge line are connected on RCS loop number 2. This base model has been verified with Maanshan Nuclear Power Station startup test data [2][3]. Figure 1 shows the whole plant scheme of Maanshan TRACE input model. Figure 2 is the detail description of major components in the input model. Plant initial condition data calculated by TRACE steady-state calculation are listed in Table 2.

Table 2 Maanshan NPS steady-state initial condition

	Plant Data	TRACE	Error (%)
Core thermal power (MW)	2822	2822	0
RCS pressure (MPa)	15.513	15.518	0.03
Total RCS flow (Mkg/hr)	49.59	49.57	0.04
Pressurizer liquid volume (m^3)	23.79	23.786	0.017
Hot-leg Temperature (K)	599.75	601.7	0.33
Cold-leg Temperature (K)	565.35	566.57	0.22
Steam generator pressure (MPa)	6.74	6.91	2.5
Steam temperature (K)	555.45	558.09	0.48
Steam generator narrow range water level (%)	50	50	0

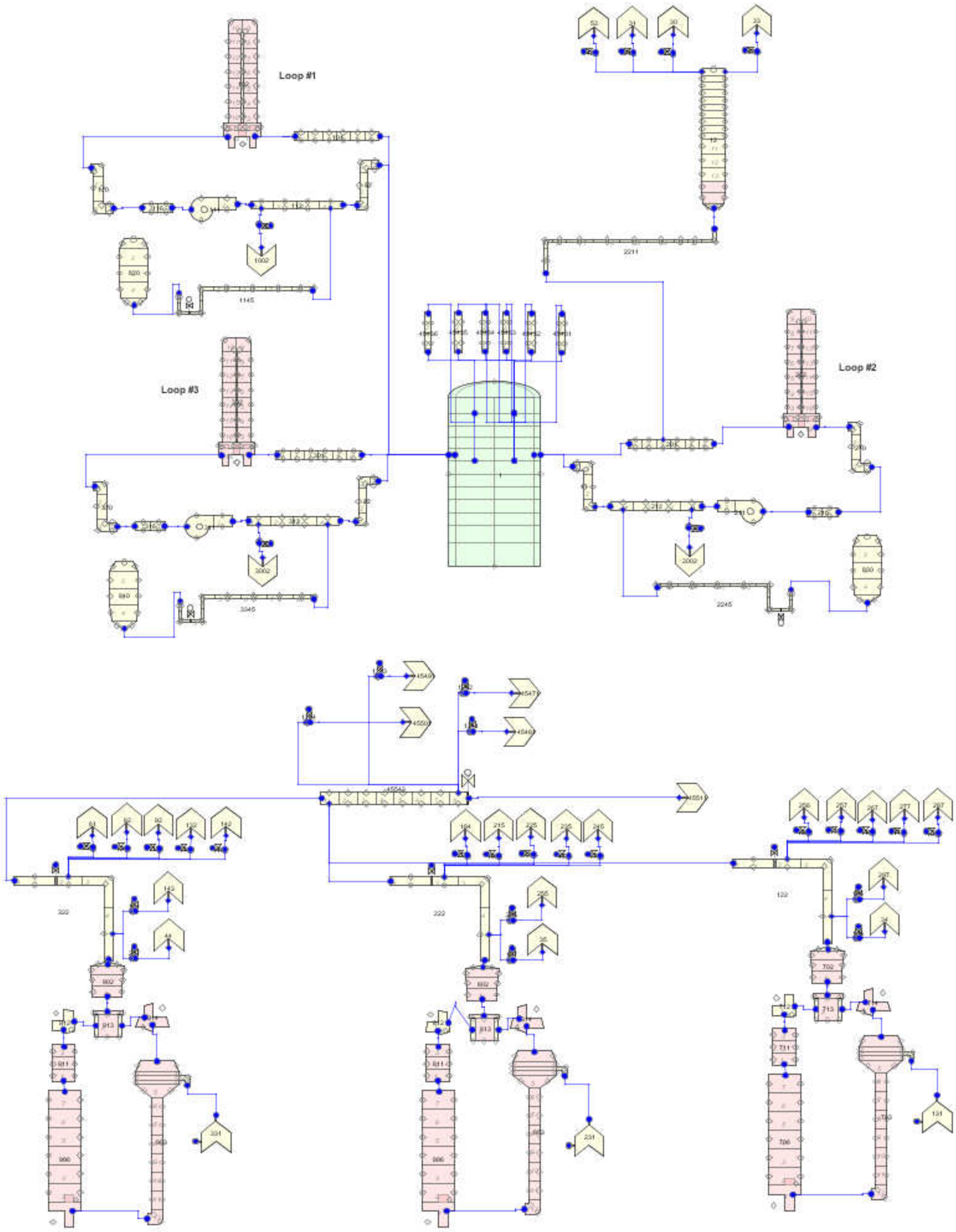

Figure 1 Schematic diagram of Maanshan TRACE input model

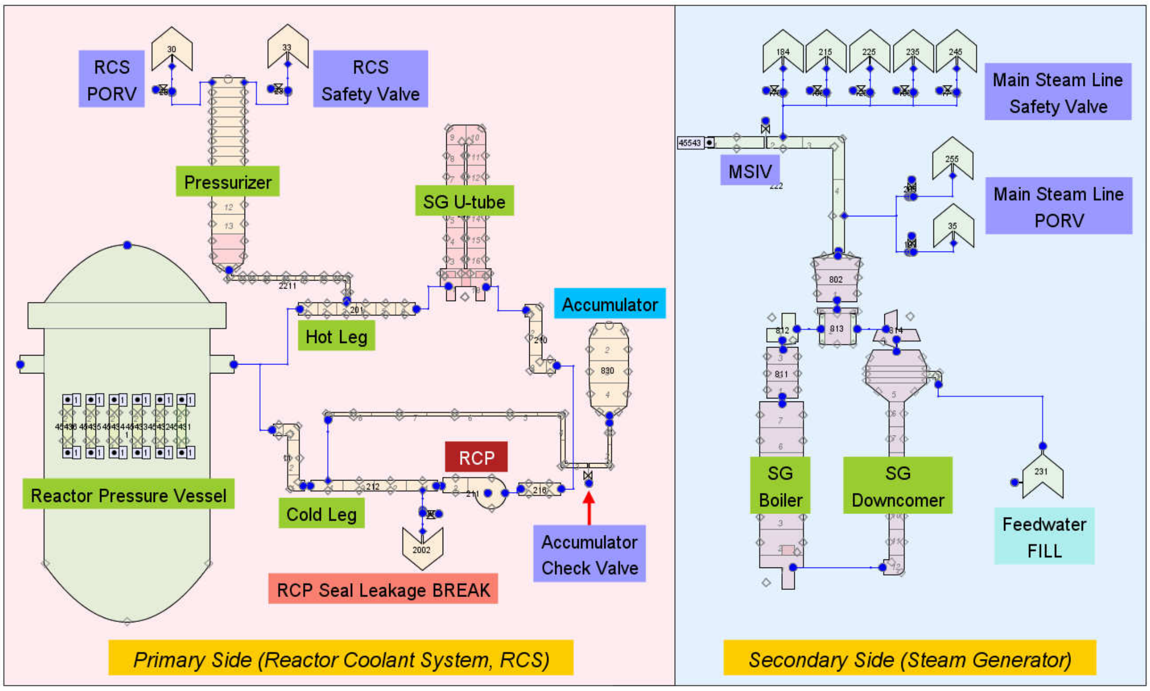

Figure 2 Detail description of major components in loop number 2

2.3 Modeling of Maanshan SBO Accident

The simulation is separated into two parts, which are (1) reaching the initial hot standby condition and (2) the simulation of SBO accident. Before the accident started, reactor was shut down and maintained at hot standby condition by operators. Steam generator water level was at about 50%, steam pressure was about 7.45 MPa, reactor coolant system pressure was 15.4 MPa, and pressurizer water level was at 62.5%. In part 1, some additional control logic in the model is used to achieve the initial condition, such as increase and control reactor coolant system pressure by pressurizer heater, controlled steam generator feedwater flow and PORV open fraction to maintain water level and steam pressure, and refill the RCS inventory by a FILL component if pressurizer water level is less than 62.5%.

In part 2, several plant data are input as boundary conditions of SBO simulation. Steam generator PORVs open fraction are controlled base on steam pressure plant data so that pressure in all three steam generators appear the same trend with plant data during the whole simulation. Steam generator auxiliary feedwater flow data are also input as a boundary condition and flow into steam generator via a FILL component connected on the steam generator downcomer, the temperature auxiliary feedwater flow is 293 K. Steam generator auxiliary feedwater input data are shown in Figure 3 to Figure 5. Reactor has shut down for about 21 hours prior to SBO accident, decay heat within the core has decreased to very low level, therefore reactor power is set constant at 0.6% (about 16.65 MW_t) during the simulation.

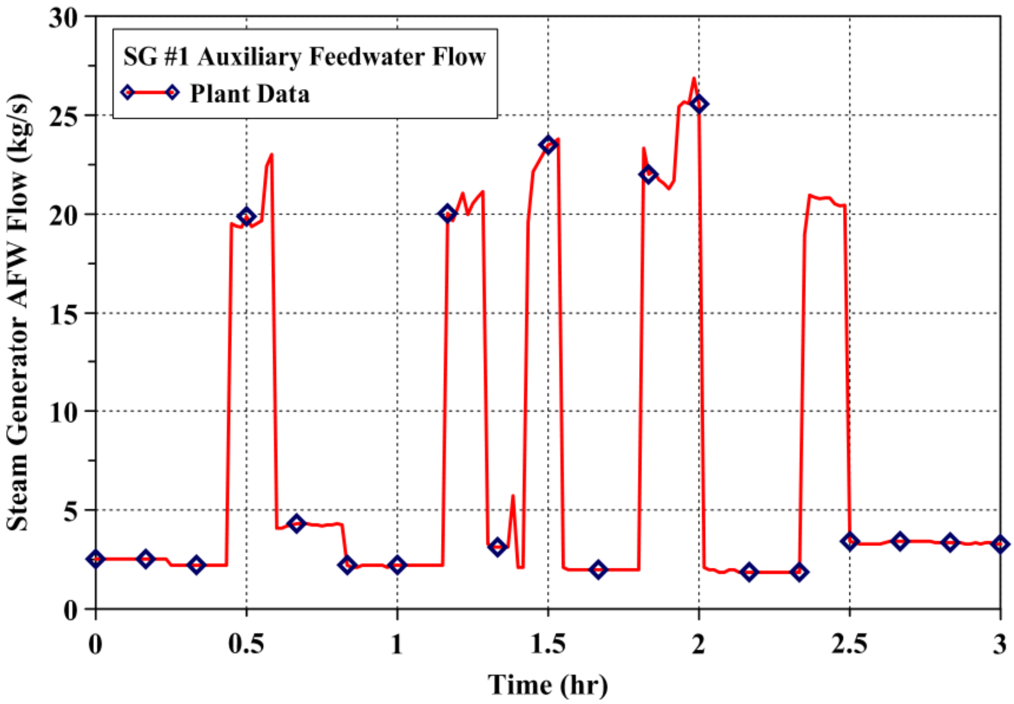

Figure 3 Input data for auxiliary feedwater flow in SG #1

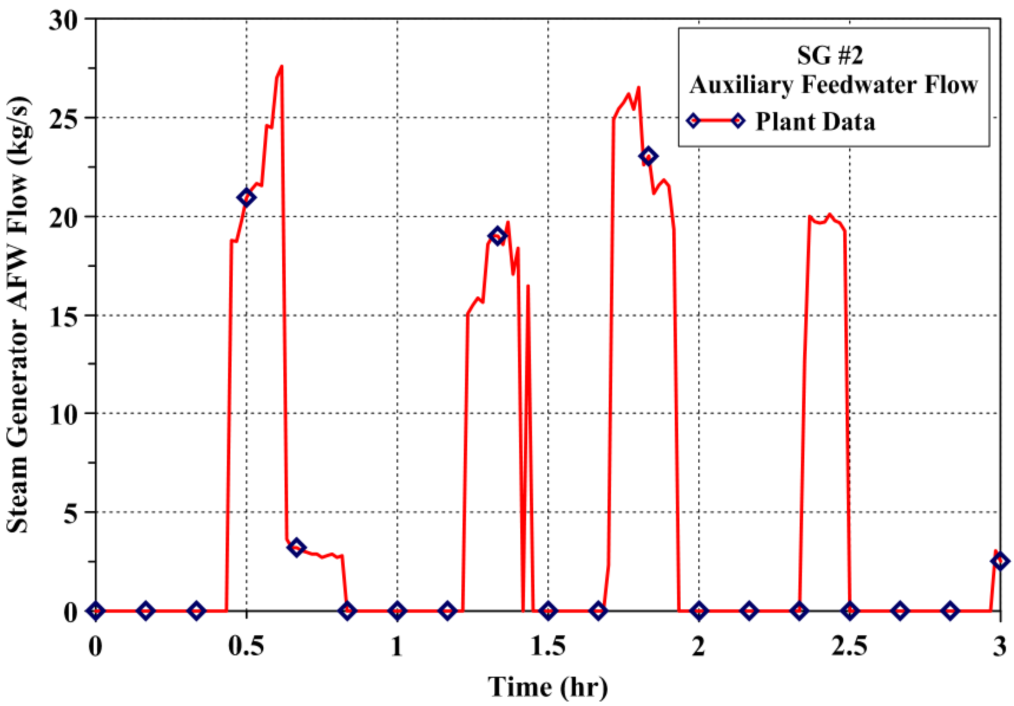

Figure 4 Input data for auxiliary feedwater flow in SG #2

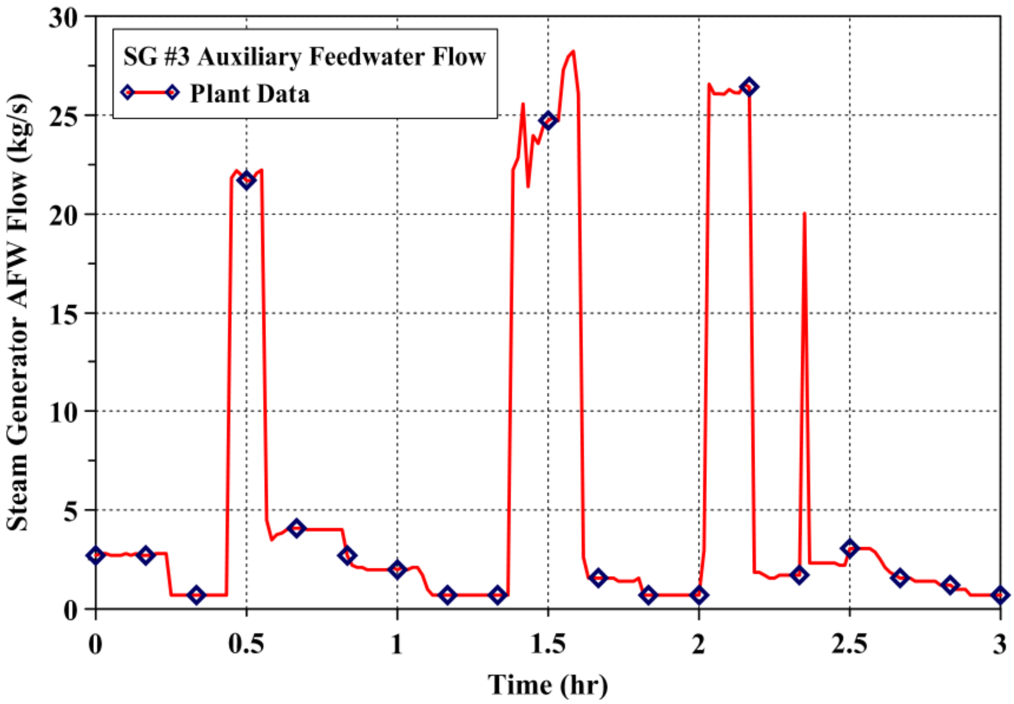

Figure 5 Input data for auxiliary feedwater flow in SG #3

2.4 <u>Simulation Results of Maanshan SBO Accident</u>

SBO happens at 16 minutes after the simulation starts. 11 minutes after SBO, turbine driven auxiliary feedwater system (TDAFW) automatically start. Operators control the auxiliary feedwater flow rate via regulating the throttling valve in order to maintain steam generator water level. The feedwater flow rate during the simulation is shown in Figure 3, 4 and 5. Due to TDAFW system, all three steam generators narrow range water level simulation results are above 50% most of the time and show similar trend with plant data. Steam generator narrow range water level results are shown in Figure 6, Figure 7, and Figure 8.

12 minutes after SBO, the operators start to initiate emergency operating procedure (EOP) to lower the steam generator pressure by opening steam line PORV. In TRACE, PORVs are controlled base on steam pressure plant data, steam generators pressure simulation results are shown in Figure 9, Figure 10, and Figure 11. Steam generator depressurization can effectively remove residual heat from reactor coolant system, therefore coolant temperature and pressure decrease as steam generator pressure become lower. Figure 12, Figure 13, and Figure 14 show the cold leg liquid temperature for three RCS loops respectively. Figure 15 shows the reactor coolant system pressure variation. As the RCS coolant temperature decrease, coolant density also becomes smaller which lead to shrinkage of RCS coolant, therefore pressurizer water level decrease. Figure 16 shows the pressurizer water level during the transient. When reactor coolant system pressure become lower than accumulator nitrogen gas pressure which is about 4.2 MPa, water inside accumulator automatically injected into RCS via two check valves.

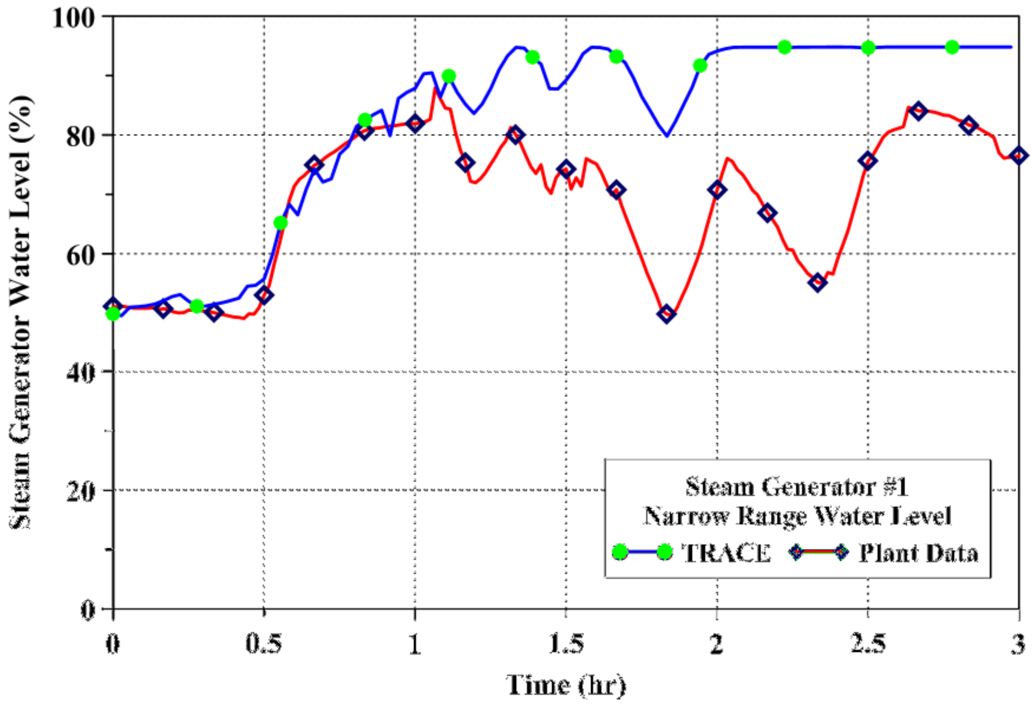

Figure 6 Steam generator #1 narrow range water level

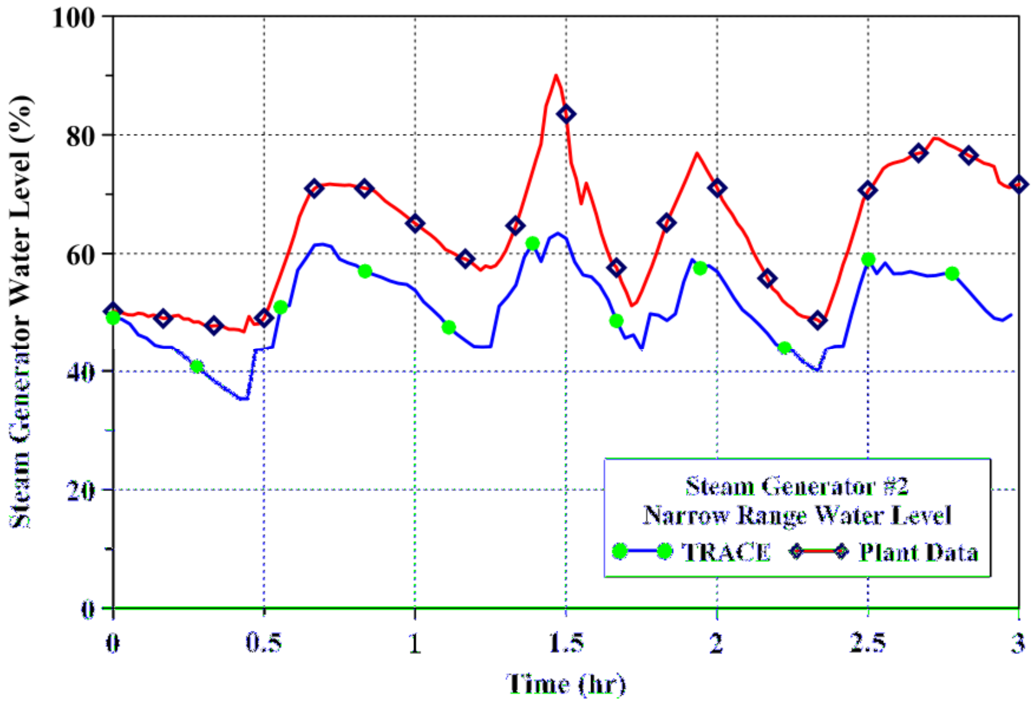

Figure 7 Steam generator #2 narrow range water level

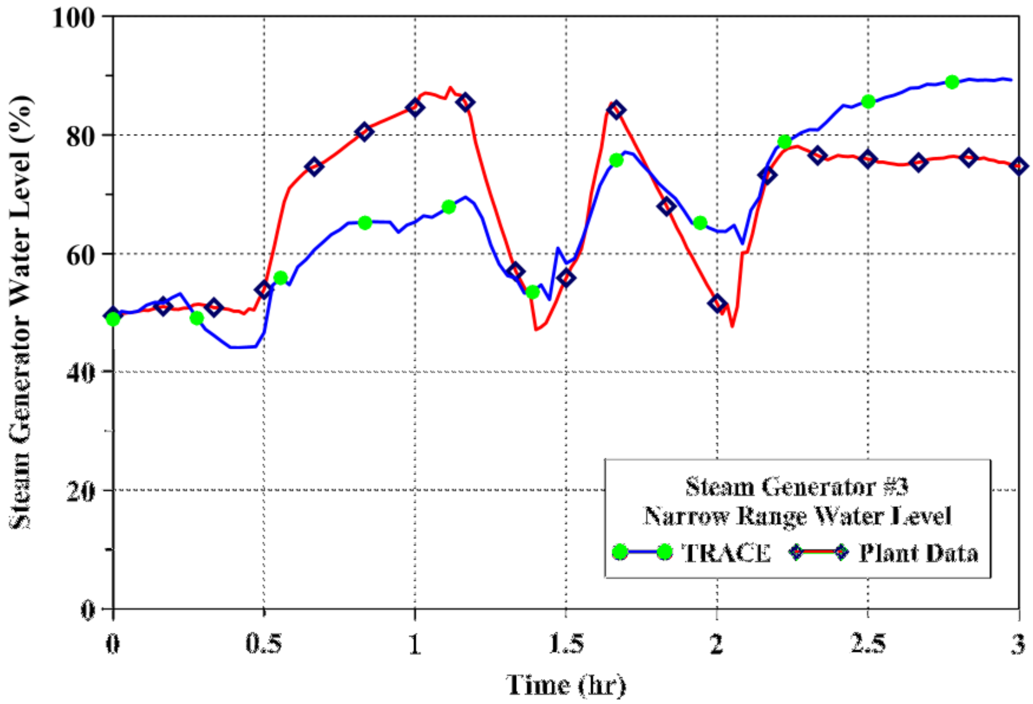

Figure 8 Steam generator #3 narrow range water level

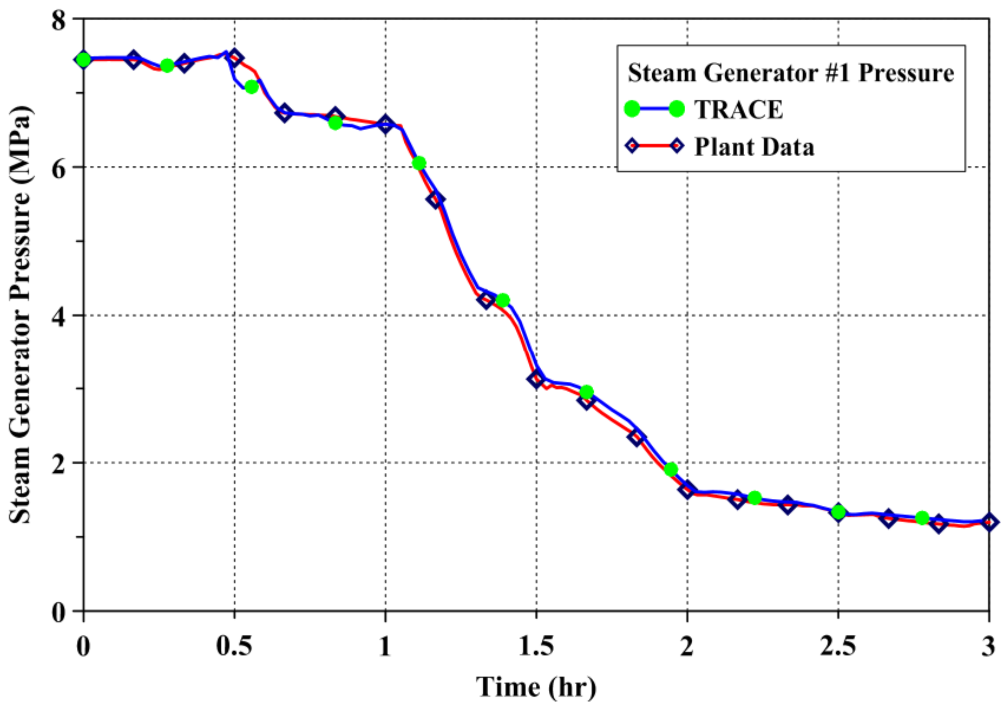

Figure 9 Steam generator #1 pressure

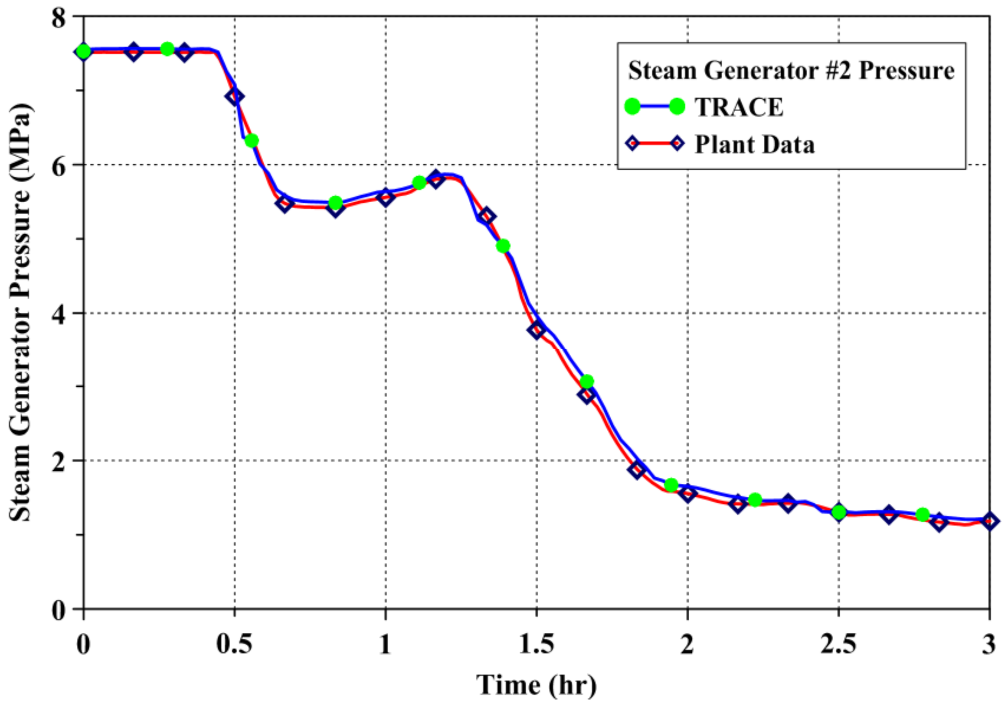

Figure 10 Steam generator #2 pressure

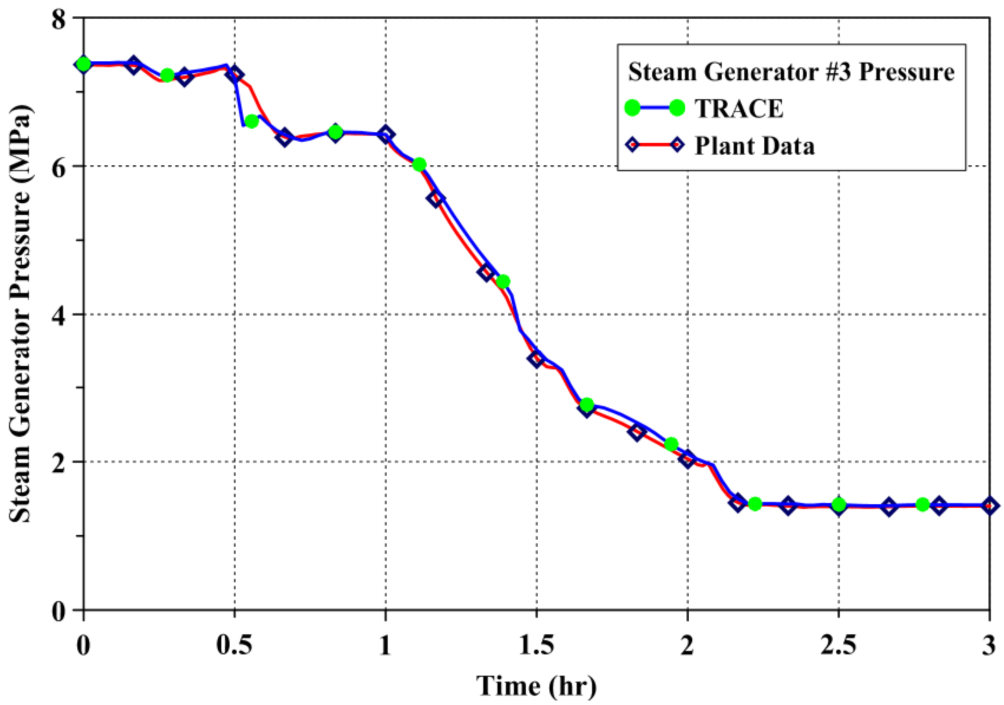

Figure 11 Steam generator #3 pressure

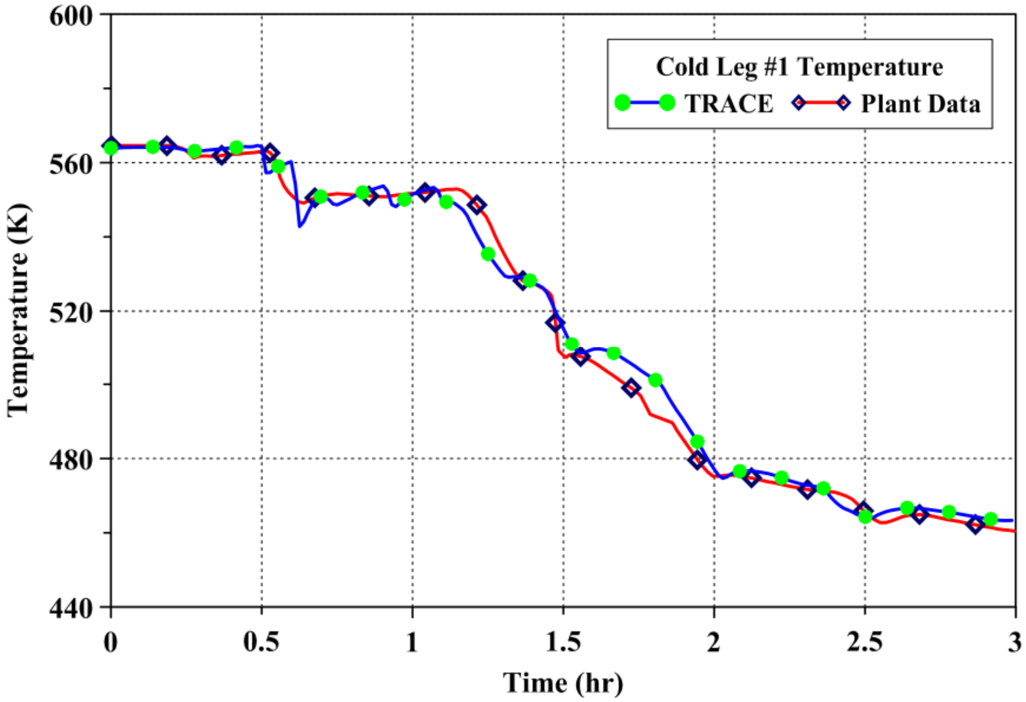

Figure 12 Cold leg #1 liquid temperature

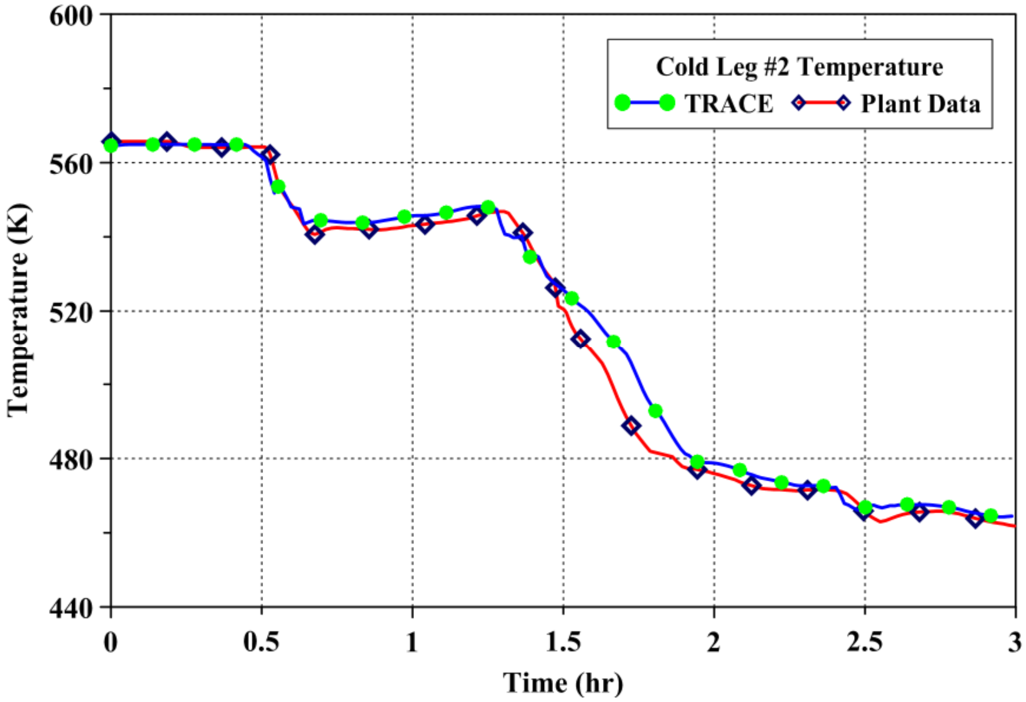

Figure 13 Cold leg #2 liquid temperature

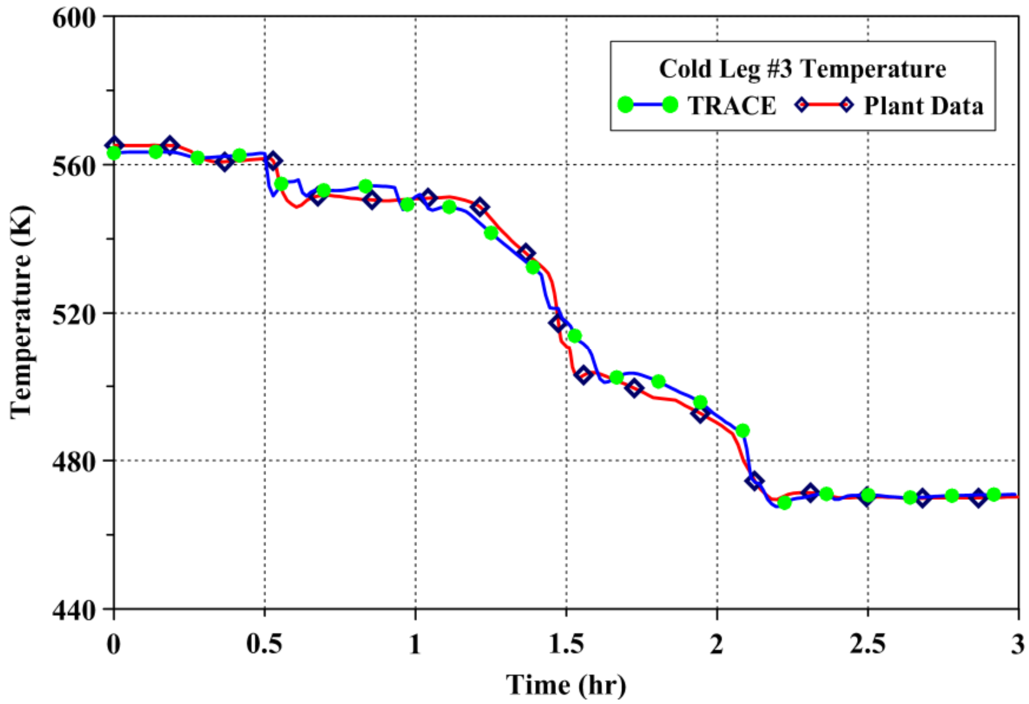

Figure 14 Cold leg #3 liquid temperature

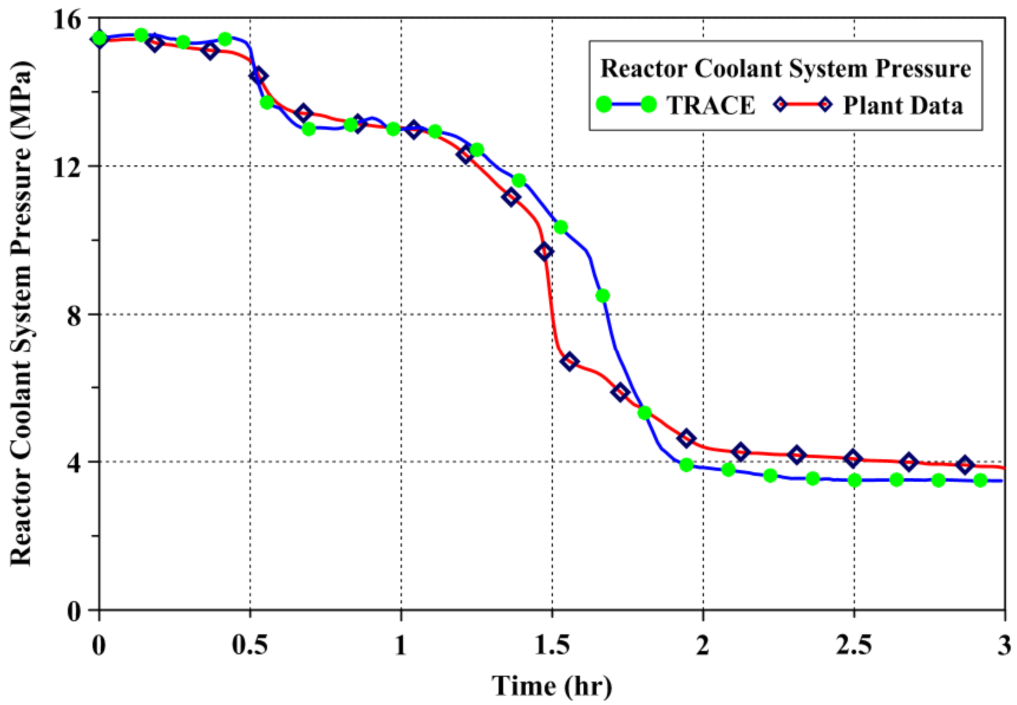

Figure 15 Reactor coolant system pressure

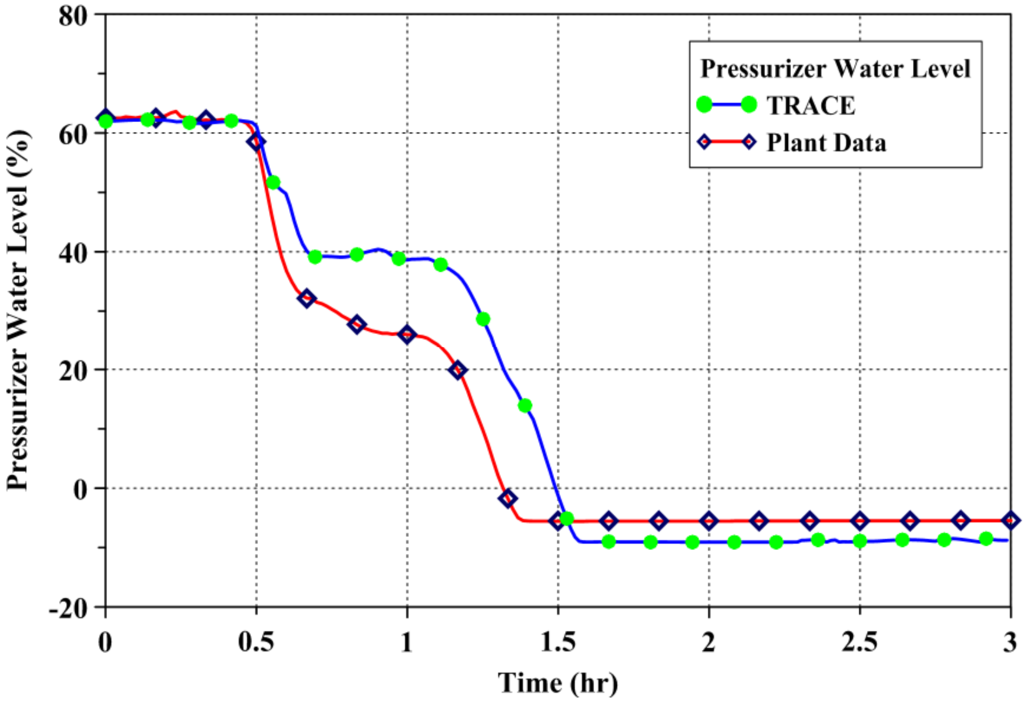

Figure 16 Pressurizer water level

2.5 Discussion

During 2 hours SBO duration, the operators successfully execute RCS cooling by controlled-depressurization of steam generators. Since no emergency power available during SBO, the turbine driven auxiliary feedwater system become the most important coolant injection system. Reactor temperature and pressure decreased form 564 K, 15.3 MPa to 472 K, 4.2 MPa respectively, and no radioactive material was released during SBO. After emergency power was recovered, residual heat removal system took place to remove the decay heat continuously. From the above results, TRACE simulation of the Maanshan SBO accident shows good agreement with the plant data. In the next chapter, several SBO mitigation strategies will be simulated by using this input model to determine the best strategy.

3. DESCRIPTION OF SBO MITIGATION STRATEGY AND MODELING METHOD

There are three base cases in this study, one without any operators' action, and the other two with different mitigation strategies during SBO. For all three cases, the plant lose the offsite power due to a large earthquake happens at 60 second, then reactor, RCP, turbine and main feedwater trip immediately. Emergency diesel generator start up automatically after losing the offsite power, the motor driven auxiliary feedwater pumps (MDAFW) then start to deliver cold feedwater into three steam generators. 20 minutes after earthquake, SBO happens and is caused by an intense tsunami that wipes out electric devices related to emergency AC power. After losing the emergency AC power, MDAFW and ECCS are not available, and turbine driven auxiliary feedwater system (TDAFW) is assumed to fail. For conservative, reactor coolant pump (RCP) seal leakage is assumed to happen in all cases. The leakage rate corresponds to full system pressure is 21 gpm (0.98 kg/s) per pump [4].

3.1 No Mitigation Strategy (Case A)

Case A has no operators' actions and is a comparison with case B and C to show the worst situation. The system response follows its default settings and logics. Calculation ends when cladding temperature reaches melting point of zirconium alloy.

3.2 Mitigation Strategy 1 (Case B)

Case B contains two main actions, including steam generators depressurization and alternate injection. After SBO happens at 20 minute with TDAFW is assumed to fail, the plant completely loss its regular coolant injection capability. Under this situation, onsite operators have to prepare any available kind of alternate injection method whatever is driven by fire truck, mobile engine driven pump, or gravity and connect these alternate equipment with the piping line in order to maintain water level in steam generators. The alternate injection preparation process is totally done by onsite operators, a minimum requirement of one hour preparation time start from tsunami wave backed away is requested by Taiwan Power Company. One hour after tsunami hit the plant, steam generator alternate injection is ready to use. Injection pressure using alternate equipment is much lower than steam generator, therefore preforming steam generator depressurization to atmospheric by opening steam line power operate relieve valve (PORV) is necessary. 0.69 MPa alternate injection operate pressure limit is used in this study, and injection flow rate and temperature are 200 gpm (12.6 kg/s) per steam generator and 20°C (293 K) respectively.

3.3 Mitigation Strategy 2 (Case C)

Case C contains three main actions, including steam generators controlled-depressurization, second stage depressurization and alternate injection. When SBO happens at 20 minute, reactor operators perform steam generator controlled-depressurization by manually adjusting the steam line PORV open fraction to maintain steam pressure at 1.57 MPa. TDAFW which is driven by steam can still operate at 1.57 MPa steam pressure, but it is assumed to fail in all cases. When alternate injection is ready at 1 hour after SBO, operators then perform second stage depressurization to depressurize steam generator to atmospheric pressure so that alternate injection system can deliver water into steam generator. Alternate injection system limitation, flow rate and temperature are the same as in case B.

3.4 Modeling of SBO Mitigation Strategy

The input model used to analyze the mitigation strategies has been verified with Maanshan startup test [2][3] and Maanshan SBO accident against plant data. Some important settings in analyzing SBO mitigation strategy are described in Table 3.

Table 3 Some important settings of SBO mitigation strategy analysis

Component / System / Control logic	Description
RCP seal leakage	Modeled by a VALVE and a BREAK. Valve flow area is adjusted to match the leak flow rate of 21 gpm (0.98 kg/s) corresponds to full system pressure.
Accumulator (ACC) and ACC isolation valve	Water temperature and N_2 pressure are 311 K and 4.24 MPa respectively. ACC will be isolated to prevent N_2 gas pressed into RCS when ACC is empty or SG pressure is lower than 0.93 MPa.
MDAFW and TDAFW	Maximum flow rate are 69.7 kg/s and 67.5 kg/s respectively for three SGs. Flow rate is adjustable to control SG water level at 50% narrow range water level.
Alternate injection	Maximum flow rate is 12.6 kg/s (200 gpm) per SG. Available when SG pressure is lower than 0.69 MPa. Flow rate control is the same as MDAFW. Alternate injection, MDAFW, and TDAFW are all connected to SG feedwater FILL.
Steam line PORV and depressurization control	Depressurization method depends on case. Depressurize SG to atmospheric pressure when SG is dryout or alternate injection ready to use.

4. SIMULATION RESULTS OF SBO MITIGATION STRATEGY

Accident starts from 60 second. Reactor trip due to large earthquake, control rods drop into reactor core therefore only decay heat remains. Decay power used in this calculation is ANS 1973 decay heat curve. RCP motor rotation speed starts to decrease and fully stopped in 200 seconds, and then natural circulation flow is established in RCS loop. After main feedwater trip at 60 second, MDAFW continue to maintain steam generator water level. 20 minutes later, MDAFW trips due to SBO. Condenser and steam dump system cannot operate under SBO.

4.1 Results of Case A

After MDAFW trip at SBO, there is no cooling water supply to steam generators. Heat from RCS continuously transfer to steam generator secondary side, steam pressure starts to rise and hold at PORV open set point of 7.96 MPa as shown in Figure 17. Steam is directly dumped into atmosphere via PORV, steam generator water level decreases slowly and dryout at about 3 hour as shown in Figure 18. RCS natural circulation flow slowly removes the decay heat to steam generator, so that RCS pressure also decreases slowly. When steam generators are all dryout, heat sink of RCS is lost. RCS natural circulation stop and pressure starts to build up. At around 3.5 hour, RCS pressure reaches and holds at pressurizer PORV open set point of 16.2 MPa, and then reactor water level starts to decrease very fast due to steam inside RCS is dumped into containment. Accumulator cannot inject cold water into RCS since RCS pressure is high, reactor water level drops to top of active fuels (TAF) at about 4.1 hour. RCS pressure and reactor water level are shown in Figure 19 and Figure 20 respectively. Without water covering fuels, peak cladding temperature (PCT) increase sharply and beyond 1088 K (1500 °F) at about 4.6 hour as shown in Figure 21.

4.2 Results of Case B

1 hour after SBO, that is, at 1.35 hour, steam generator alternate injection is ready, operators then open the steam line PORV to depressurize steam generator to atmospheric pressure. Steam generator water level decrease and become dryout at about 2 hour due to rapid depressurization. When steam generator pressure is lower than 0.69 MPa at around 2.1 hour, alternate injection starts to inject cold water and steam generator water level starts increasing. RCS pressure also decrease sharply at 1.35 hour, it's because rapid steam generator depressurization can remove massive heat via heat transfer between RCS and SG secondary side. Reactor water level decrease at 1.35 hour is mainly because RCS water density become smaller caused by decreasing of RCS temperature, so that RCS water volume shrink and cause the water level to decrease. Accumulator can inject cold water to RCS when RCS pressure is lower than 4.2 MPa but isolated when steam generator pressure is lower than 0.93 MPa. Steam generator water level is recovered to normal position at about 4.3 hour. Reactor water level is finally stabilized at 1.54 m above TAF, and PCT is well below 1088 K (1500 °F).

4.3 Results of Case C

At 20 minute when SBO happen, operators perform steam generator controlled-depressurization by opening steam line PORV and hold the steam pressure at 1.57 MPa as shown in Figure 17 until alternate injection is ready. Since steam pressure is under controlled, less steam is dumped into atmosphere so that steam generator water level decrease slower in controlled stage and doesn't become dryout. At 1.35 hour, operators perform second stage depressurization to depressurize steam generator to atmospheric pressure so that alternate injection can come into steam generator. RCS pressure decrease rapidly and stay at around 2.5 MPa is due to controlled-depressurization of steam generator. Steam generator water level is recovered to normal position at about 4 hour. Injection flow rate needed to maintain steam generator water level at normal position after 4 hour is about 8 kg/s for three SGs. Reactor water level is finally stabilized at 1.84 m above TAF, and PCT is well below 1088 K (1500 °F).

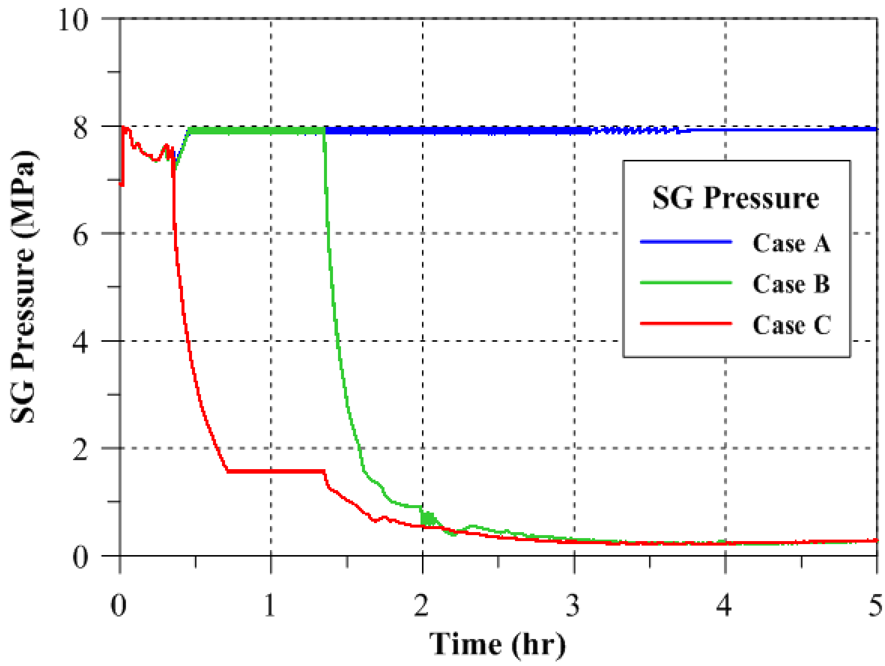

Figure 17 Steam generator pressure for cases A, B, and C

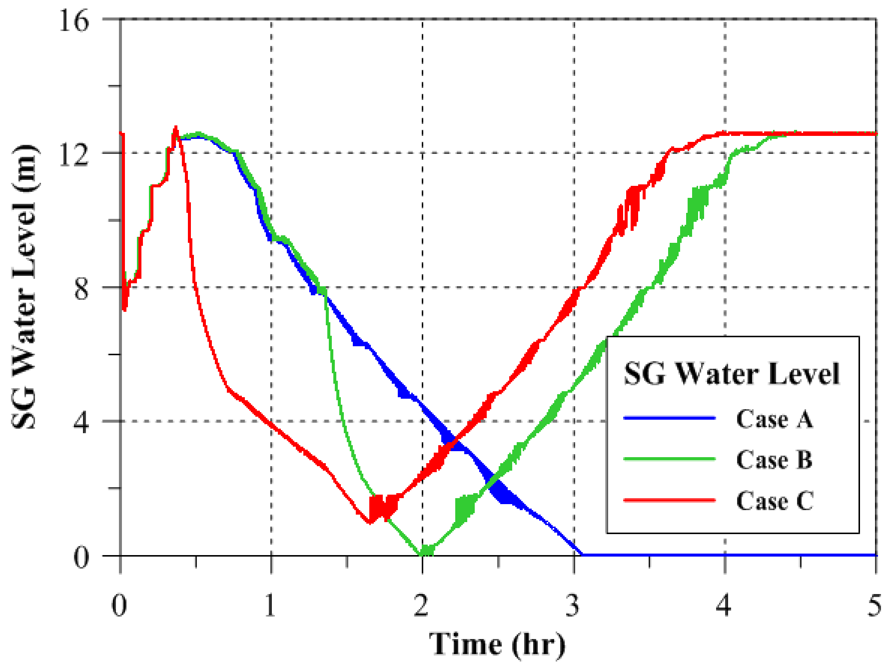

Figure 18 Steam generator water level for cases A, B, and C

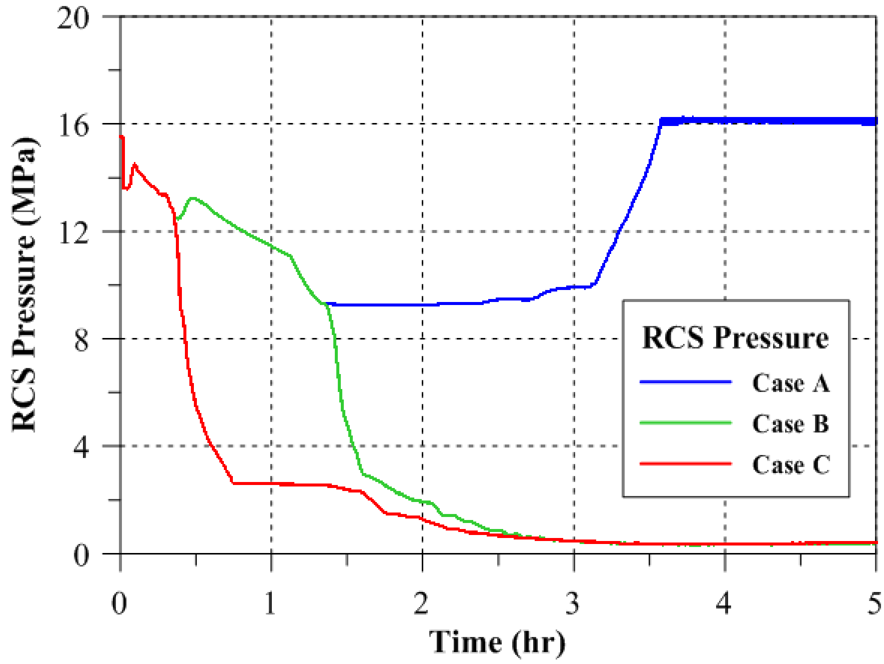

Figure 19 Reactor coolant system pressure for cases A, B, and C

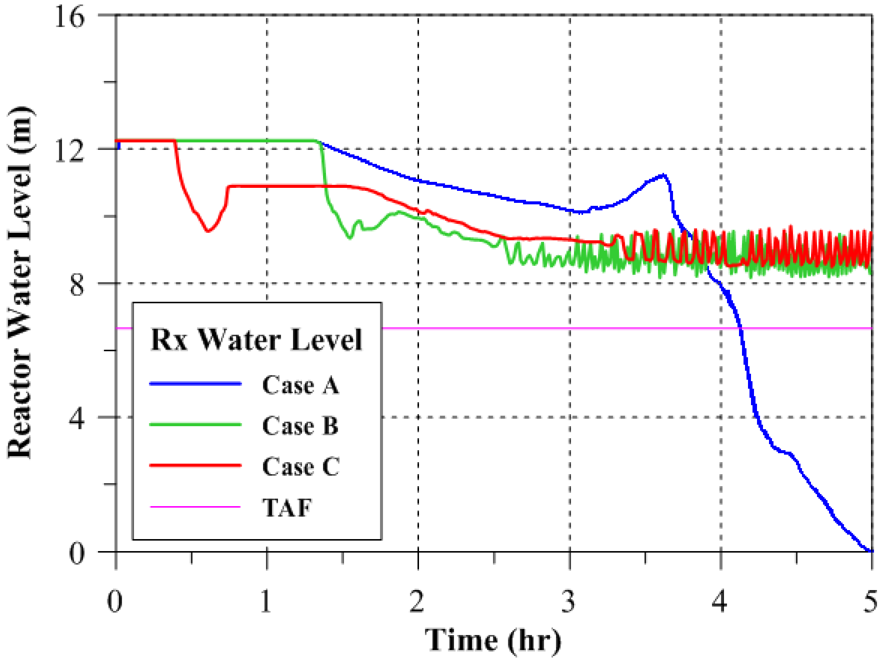

Figure 20 Reactor water level for cases A, B, and C

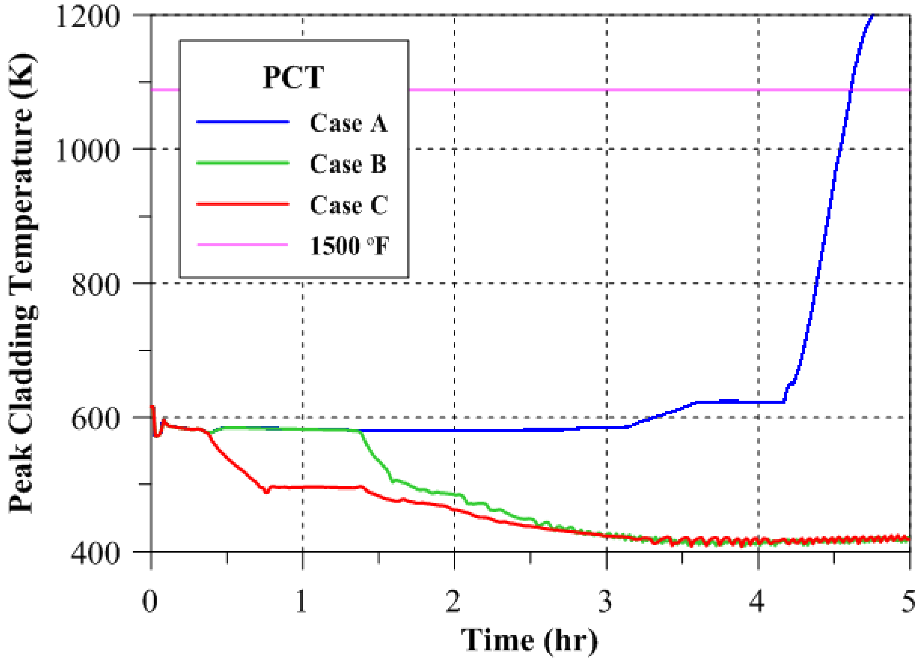

Figure 21 Peak cladding temperature for cases A, B, and C

4.4 <u>Discussion</u>

The result from case A shows that core damage will occur within 5 hours after the earthquake if nothing has been done. Recall that the mitigation strategy is used in this kind of accident to prevent PCT from exceeding 1088 K (1500 °F) which is the temperature that metal-water reaction can self-sustain. If a large amount of hydrogen is generated by metal-water reaction, hydrogen explosion may occur and further compromise the integrity of reactor or containment. Results of case B and C show that both two strategies successfully keep the fuels covered with water and PCT is not higher than 1088 K. The benefit of depressurizing steam generator is indirectly remove the decay heat from RCS vie steam generator without losing RCS inventory, but the steam generator water level decrease. From Figure 18, it's obvious that steam generator water level decrease rapidly during depressurization, water level decreasing rate is even faster in case B and become dryout before alternate injection flow can come into steam generator. Controlled-depressurization in case C shows that not only RCS temperature and pressure can be reduced in the early stage of accident but also keep the steam generator from being dryout. In addition, RCP seal leakage flow decrease with decreasing RCS pressure, reducing the inventory loss in RCS. Therefore, strategy in case C is recommended for coping with SBO. Several studies of this strategy have also been done, the assumptions may be different but system responses are very similar [5].

5. SENSITIVITY STUDY ON ALTERNATE INJECTION TIME

When preforming strategy in case C, alternate injection preparation time is a crucial parameter to guarantee the success. If the preparation time is too long, steam generator will become dryout and RCS pressure will build up to pressurizer PORV open set point. RCS inventory loss due to steam that dumped into containment via pressurizer PORV is not recoverable since ECCS is not available, although accumulator can achieve passive injection but the water volume in the tank is limited. 10 different injection times have been tested. For cases without TDAFW, injection ready at 2, 3, 3.5, 4, and 4.5 hours after SBO are named as case C1, C2, C3, C4, and C5 respectively. For cases with TDAFW operates 30 minutes after SBO, injection ready at 4, 6, 7, 8, and 9 hours after SBO are named as case C6, C7, C8, C9, and C10 respectively. Alternate injection flow rate for cases C1-C10 are all 200 gpm (12.6 kg/s) per steam generator.

5.1 Cases without Turbine Driven Auxiliary Feedwater

The results show that RCS pressure will raise to pressurizer PORV open set point if steam generator alternate injection preparation time is later than 3 hours. For case C3 with 3.5 hours preparation time, RCS pressure re-decrease when heat transfer to steam generator is reestablish, and reactor water level is stabilized at about 1.35 m above TAF. Therefore, the alternate injection preparation time should be no later than 3.5 hours after SBO to ensure fuels are covered with water. RCS pressure, reactor water level, and PCT for cases C1-C5 are shown in Figure 22, Figure 23, and Figure 24 respectively.

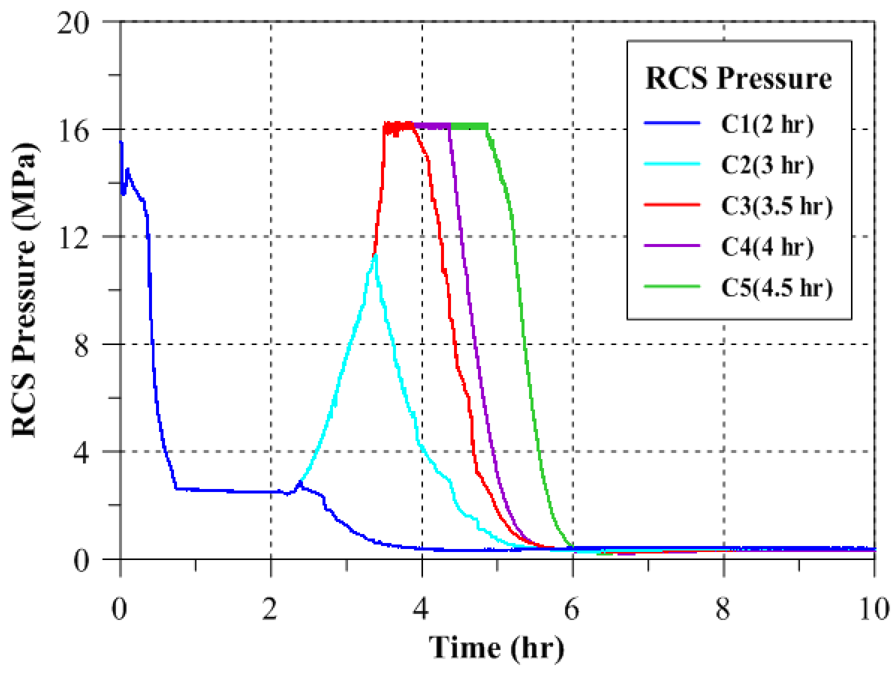

Figure 22 Reactor coolant system pressure for cases C1-C5

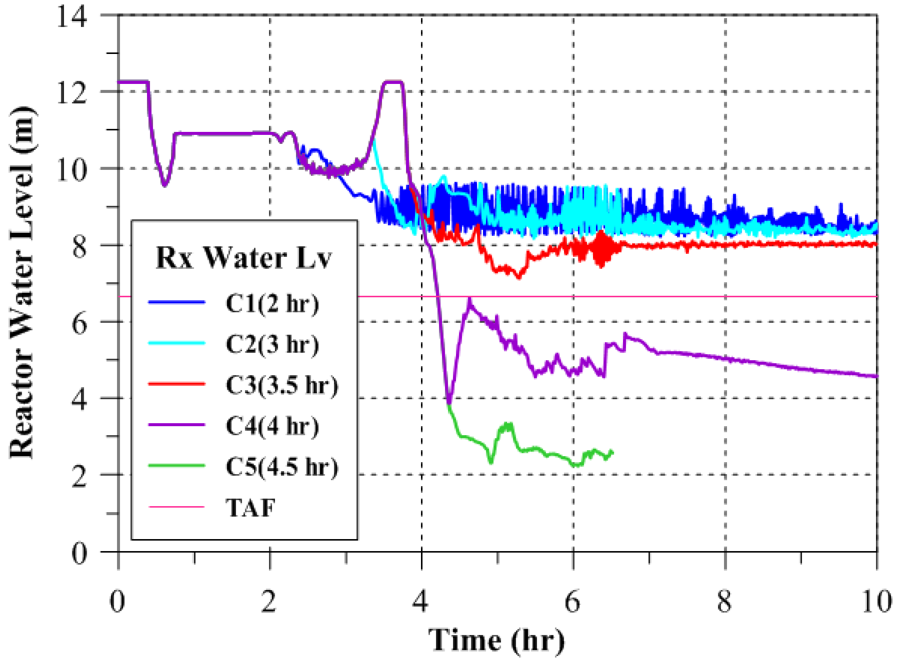

Figure 23 Reactor water level for cases C1-C5

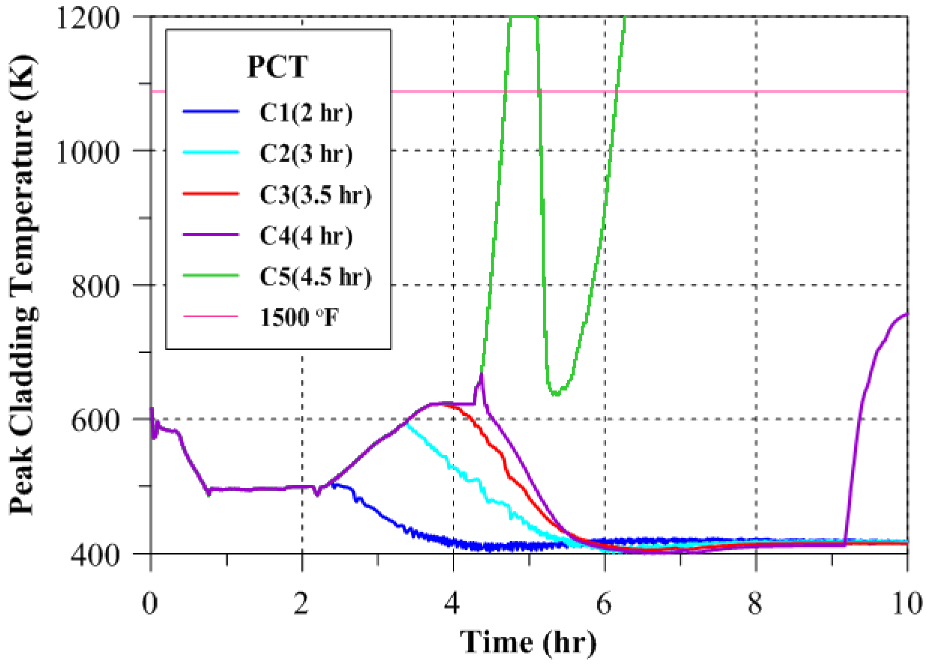

Figure 24 Peak cladding temperature for cases C1-C5

5.2 Cases with Turbine Driven Auxiliary Feedwater Operating for 30 Minutes

If TDAFW operates for 30 minutes after SBO, steam generator alternate injection preparation time can be further extended because auxiliary feedwater is supplied while steam generator is performing controlled-depressurization. The results show that, to prevent pressurizer PORV from opening, preparation time should be no more than 7 hours after SBO. Results of reactor water also show that preparation time that greater than 7 hours will lead to core uncovery. Therefore, if TDAFW operates 30 minutes after SBO, the alternate injection preparation time should be no later than 7 hours after SBO to ensure fuels are covered with water. RCS pressure, reactor water level, and PCT for cases C6-C10 are shown in Figure 25, Figure 26, and Figure 27 respectively.

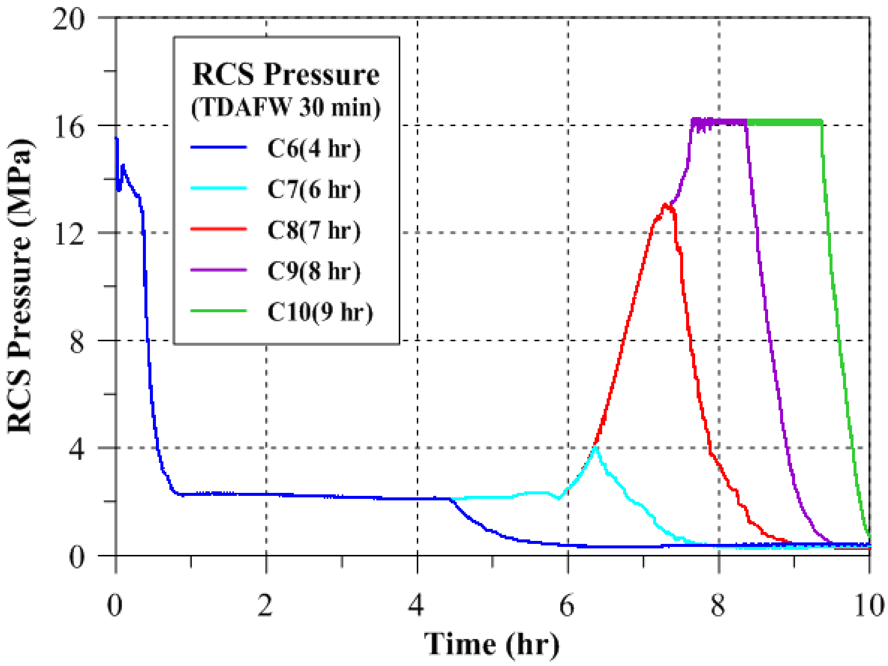

Figure 25 Reactor coolant system pressure for cases C6-C10

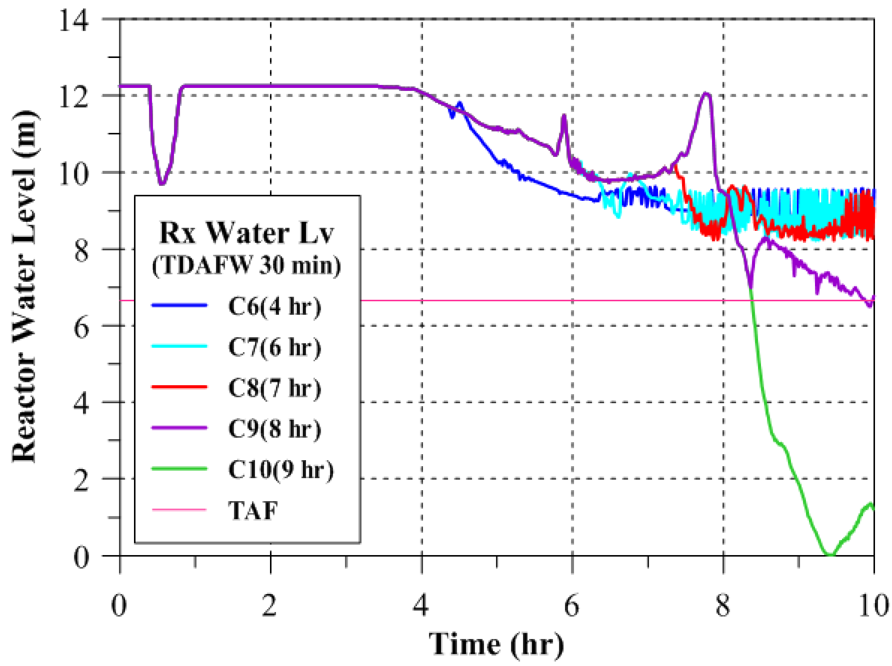

Figure 26 Reactor water level for cases C6-C10

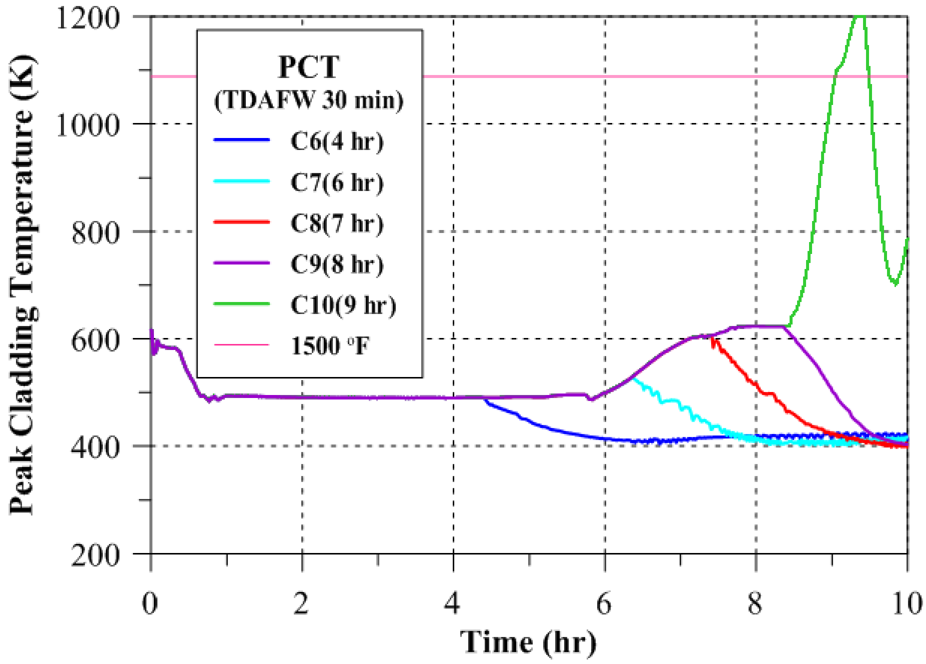

Figure 27 Peak cladding temperature for cases C6-C10

6. EFFECT OF RCP SEAL LEAKAGE

RCP seal cooling is lost after loss of offsite power that cause the degradation of pump shaft seal components. RCS inventory loss due to RCP seal leakage is not recoverable until AC power is recover. RCP leakage rate depends on combinations of failure of the seal components, the effect of RCP seal leakage is performed in this research by testing two leakage rates which are 21 gpm (0.98 kg/s) and 182 gpm (8.5 kg/s) per RCP. According to the references [4][6], the occurrence probability of 21 gpm leak rate after 13 minutes is 0.79 and the occurrence probability of 182 gpm leak rate after 13 minutes is 0.1975. The base scenario in this section is identical to case C6, except the steam generator alternate injection is ready at 1 hour after SBO. The result shows that for 182 gpm leak rate, core uncover happens at 8 hours after SBO. For 21 gpm leak rate, which has the highest occurrence probability, core uncover time can be further extended to 75 hours after SBO. RCP seal leakage rate, integrated leakage rate and reactor water level are shown in Figure 28, Figure 29 and Figure 30 respectively.

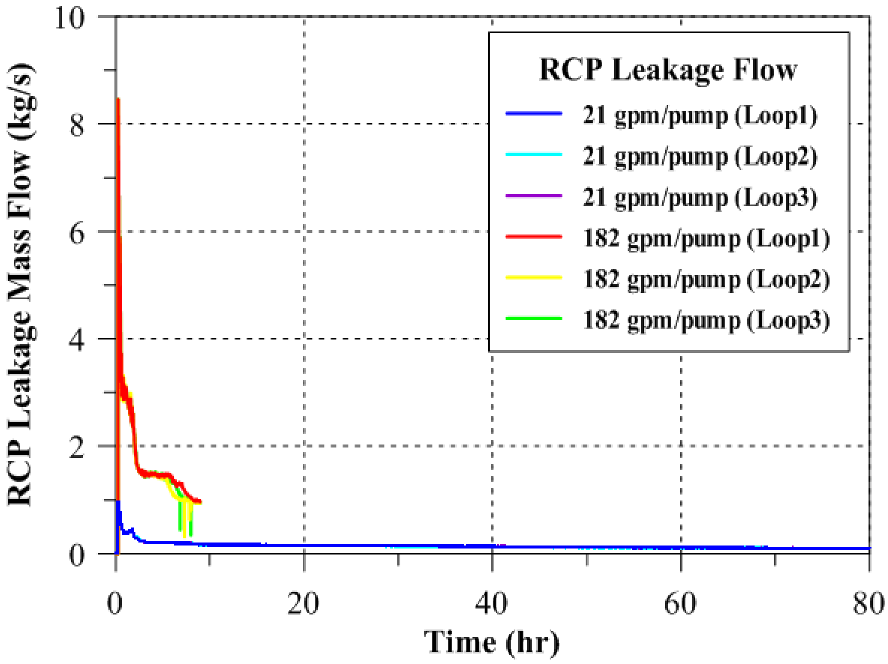

Figure 28 RCP seal leakage flow

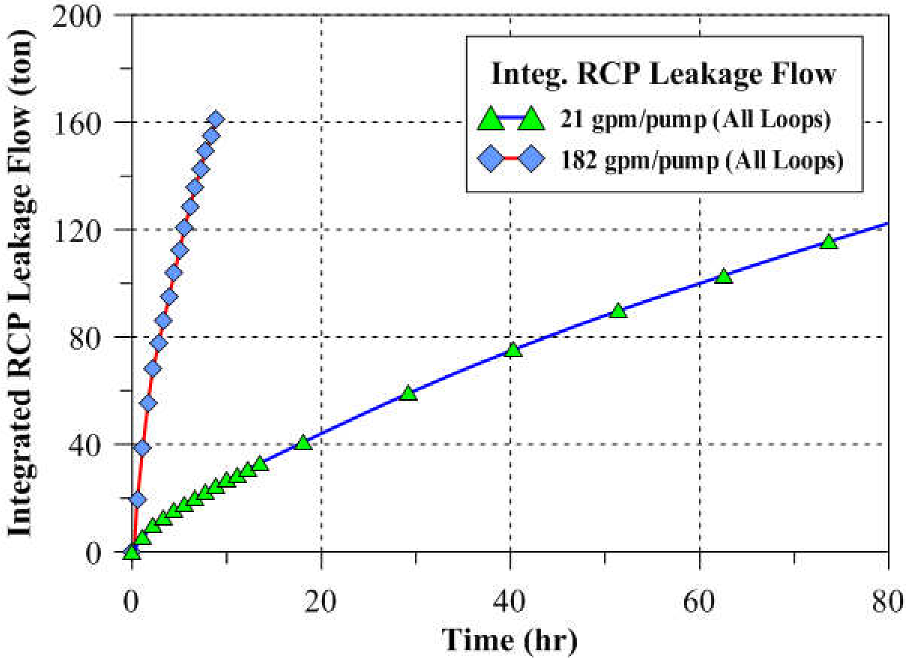

Figure 29   Integrated RCP seal leakage flow

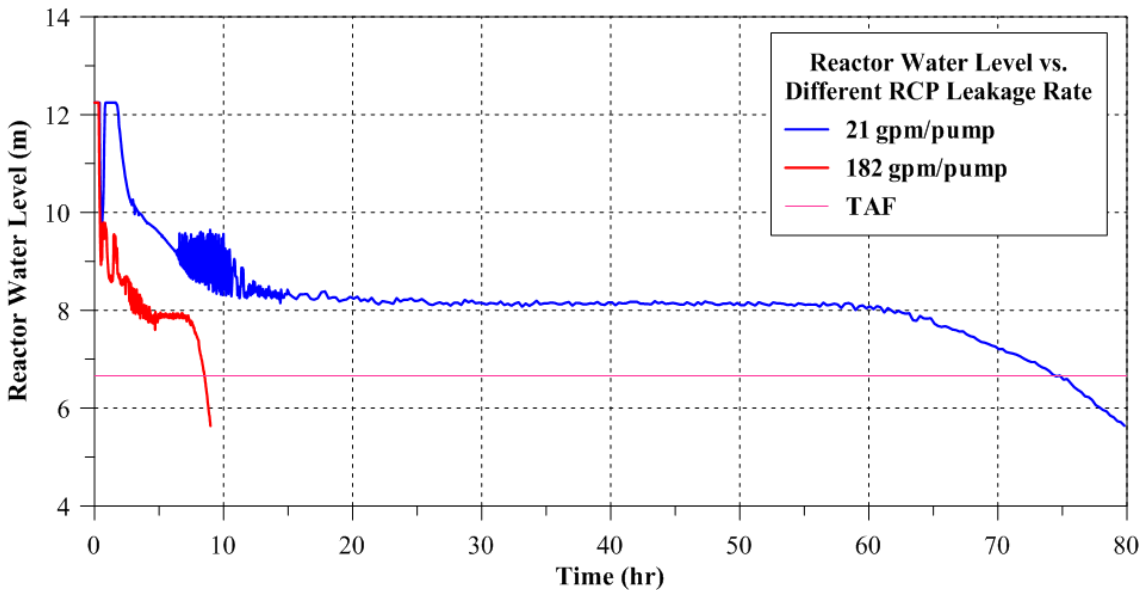

Figure 30   Reactor water level vs. different RCP seal leakage rate

7. CONCLUSIONS

When facing beyond design basis accidents, great uncertainties are associated with regular plant systems and components. Therefore, the strategy that can bring the plant to safety condition as soon as possible should be considered. The simulation results in this research show that performing steam generator controlled-depressurization at the early stage of accident and, if no regular coolant injection system available, line up the alternate injection system in 3.5 hours after SBO can keep the fuels covered with water. In addition, after plant is under controlled at the early stage of accident, onsite operators should recover AC power as quickly as possible so that ECCS can make up RCS inventory loss through RCP seal, and residual heat removal system (RHR) can remove the decay heat in RCS continuously until reactor is at cold shut down.

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10. SUPPLEMENTARY NOTES
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11. ABSTRACT *(200 words or less)*

Maanshan Nuclear Power Station is a two-unit Westinghouse three-loop PWR power station. This research studies the simulation of Maanshan SBO accident happened on 18th March, 2001, and thermal-hydraulic phenomena of the plant during station blackout with and without mitigation strategies. The modeling and simulation works were done by using TRACE code, which is a best-estimate thermal-hydraulic system code developed by US NRC. The purpose of using the mitigation strategy during SBO is to cool down NSSS as soon as possible, to keep the fuel covered by water, and not to let peak cladding temperature (PCT) higher than 1088K (1500°F), which is the temperature that metal-water reaction can self-sustain. Actions that considered such as operation of auxiliary feedwater system, depressurization of steam generators (SG), and line-up the alternate water sources such as sea water when regular systems aren't available.

The simulations of mitigation strategies start from normal operation at 100% power then an earthquake is assumed to happen, tsunami strike the site 20 minutes later, and failure of turbine driven auxiliary feedwater is assumed in base cases. Two different basic mitigation strategies been simulated, include (1) SG controlled-depressurization at the time that SBO happen with 1 hour SG alternate injection preparation time, (2) SG alternate injection ready in 1 hour, SG depressurization at the time that injection is ready. In addition, alternate injection preparation time is further extended to find the longest acceptable value. Reactor coolant pump shaft seal leakage is also considered in all cases.

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