NUREG/IA-0496

International Agreement Report

The Analysis and Study of ELAP Event and Mitigation Strategies Using TRACE Code for Maanshan PWR

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Washington, DC 20555-0001

Manuscript Completed: January 2018
Date Published: February 2019

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

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ABSTRACT

In this study, TRACE code was used to evaluate the postulated Extended Loss of AC Power (ELAP) accident in Maanshan nuclear power plant (NPP), determining whether RCS water level will drop down below Top of Active Fuel (TAF) while the 5th diesel generator and gas turbines are all disabled when the accident occurred. The scenario and assumptions of postulated ELAP event in this study were referred to the WCAP-17601-P report and NUREG-1953 report. To analyze the effectiveness of URG and FLEX strategies, this research will run a base case without any mitigation strategy and four cases with multiple mitigation strategy under different conditions. According to the results of simulation, it can be found that all four cases in this study can keep RCS water level above TAF, ensuring the safety function of reactor.

FOREWORD

The U.S. NRC (United States Nuclear Regulatory Commission) has developed a thermal hydraulic analysis code-RELAP5. RELAP5 has been designed to perform best-estimate analysis of loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in reactor systems. Models used include multidimensional two-phase flow, non-equilibrium thermo-dynamics, generalized heat transfer, reflood, level tracking, and reactor kinetics. Traditionally, the RELAP5 code analysis model was developed by ASCII file, which was not intelligible for the beginners of computer analysis. Fortunately, a graphic input interface, SNAP (Symbolic Nuclear Analysis Program) is developed by Applied Technology Incorporation Inc. and the model development process becomes more conveniently.

To obtain the authorization of these 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 code. NTHU (National Tsing Hua University) is the organization in Taiwan responsible for the application TRACE and SNAP in thermal hydraulic safety analysis. To meet this responsibility, the TRACE/SNAP model of Maanshan nuclear power plant has been developed. This model was used to perform the URG and FLEX strategies study for ELAP.

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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. NTHU 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 NTHU.

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.

SNAP is an interface of NPP analysis codes and developed by U.S. NRC and Applied Programming Technology, Inc. Different from the traditional input deck in ASCII files, the graphical control blocks and thermal hydraulic connections make researches comprehend the whole power plant and control system more easily. Due to these advantages, the RELAP5/MOD3.3 model for Lungmen NPP was established with SNAP interface in this research.

The Maanshan NPP operated by Taiwan Power Company (TPC) is the only Westinghouse-PWR in Taiwan. The rated core thermal power is 2775 MW. The reactor coolant system has three loops, each of which includes a reactor coolant pump and a steam generator. The pressurizer is connected to the hot-leg piping in loop 2. 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 model is used to analyze the mitigation strategies.

Then, the TRACE/SNAP model was used to evaluate the postulated (ELAP) accident in Maanshan NPP, determining the effectiveness of URG and FLEX. According to the results of simulation, it can be found that the cases with the URG or FLEX can keep RCS water level above TAF, ensuring the safety function of reactor.

ABBREVIATIONS AND ACRONYMS

ACC	Accumulator
BDBEE	Beyond-Design-Basis External Event
CAMP	Code Applications and Maintenance Program
ELAP	Extended Loss of AC Power
EOP	Emergency Operating Procedure
FLEX	Flexible and Diverse Coping Strategies
FSAR	Final Safety Analysis Report
LOCA	Loss-Of-Coolant Accident
NPP	Nuclear Power Plant
NTHU	National Tsing Hua University
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
PCT	Peak Cladding Temperature
SBO	Station Blackout
SG	Steam Generator
SNAP	Symbolic Nuclear Analysis Program
TAF	Top of Active Fuel
TDAFW	Turbine Driven Auxiliary Feedwater System
TPC	Taiwan Power Company
URG	Ultimate Response Guideline
U.S. NRC	United States Nuclear Regulatory Commission

1 INTRODUCTION

On March 11th 2011, an intense earthquake and tsunami hit Fukushima Daiichi power station, and made a severe damage to the nuclear power plant (NPP). This tragedy shows that beyond design basis external event (BDBEE) may let power station falls into certain circumstances such as loss of ultimate heat sink and loss of alternating AC power, which also called station blackout (SBO). Both of these events are considered as the most severe situation to the nuclear power plant. Fukushima accident demonstrates that without proper emergency equipment or mitigation strategy, NPP may in risk while facing such similar situation. To cope with such BDB event, Taiwan Power Company (TPC) has developed a method called Ultimate Response Guidelines (URG) [1], which gives operators the rights taking emergency steps to avoid the reactor core melting or the hydrogen accumulation inside the containment. Once either AC power or water supply cannot be restored in time, or there's an earthquake and tsunami larger than safe shutdown, URG will be activated. The main action of URG including 2 steps depressurization, alternative water injection and containment venting. In addition, U.S. NRC also developed a mitigation strategy called Diverse and Flexible Coping Strategy (FLEX) to tackle with such extended SBO condition [2]. The main purpose of FLEX was to support key safety functions by providing multiple means of power and water supply, which can mitigate the consequence of beyond design basis external event.

After Fukushima accident, from many investigative reports [3]-[4], SBO situation may be longer than we have concerned. The emergency response guidelines of severe accident should be modified to cope with such extended loss of AC power (ELAP) event. In this study, TRACE code was used to evaluate the postulated ELAP accident in Maanshan nuclear power plant, determining the effectiveness of URG and FLEX. TRACE was developed by U.S. NRC [5], which is for NPP thermal hydraulic analysis, and usually applied to analyze the transients or accidents data [6]-[at nuclear power plant. In addition, Maashan TRACE/SNAP model has been built in previous study. Maanshan nuclear power plant is a 2-units Westinghouse 3-loops PWR power station operated by Taiwan power company since 1984. The accuracy and availability of Maanshan NPP TRACE/SNAP model has been verified by the comparison between FSAR data and Maanshan startup test 9].

This research first analyzes the Maanshan SBO accident happened on 18 March, 2001 by gusing Maanshan NPP TRACE/SNAP model and compares the results with plant data to confirm the accuracy of TRACE/SNAP model again. Then, the study and analysis for ELAP were performed in this study. The scenario and assumptions of postulated ELAP event in this study were referred to the WCAP-17601-P report [10] and NUREG-1953 report [11], determining whether RCS water level will drop down below TAF while the 5th diesel generator and gas turbines are all disabled when the accident occurred. To analyze the effectiveness of URG and FLEX strategies, this research will run a base case without any mitiation strategy and four cases with multiple mitigation strategies under different conditions.

2 VERIFICATION OF MAANSHAN TRACE/SNAP MODEL FOR THE SBO EVENT

2.1 Introduction of Maanshan NPP SBO Event

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 0:57am, 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

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 Maanshan TRACE/SNAP model and the simulation starts from 0:30 am to 3:30 am, March 18th.

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 The Description of Maanshan NPP TRACE/SNAP Model

The computer code used in this research is TRACE V5.0p4 and the model is edited by using SNAP V2.5.1. Maanshan TRACE base model contains 136 hydraulic components, 682 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 NPP startup test and FSAR data. Figure 1 shows the whole plant scheme of Maanshan TRACE/SNAP model. Plant initial condition data calculated by TRACE steady-state calculation are listed in Table 2.

Table 2 Maanshan NPP 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 (m3)	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

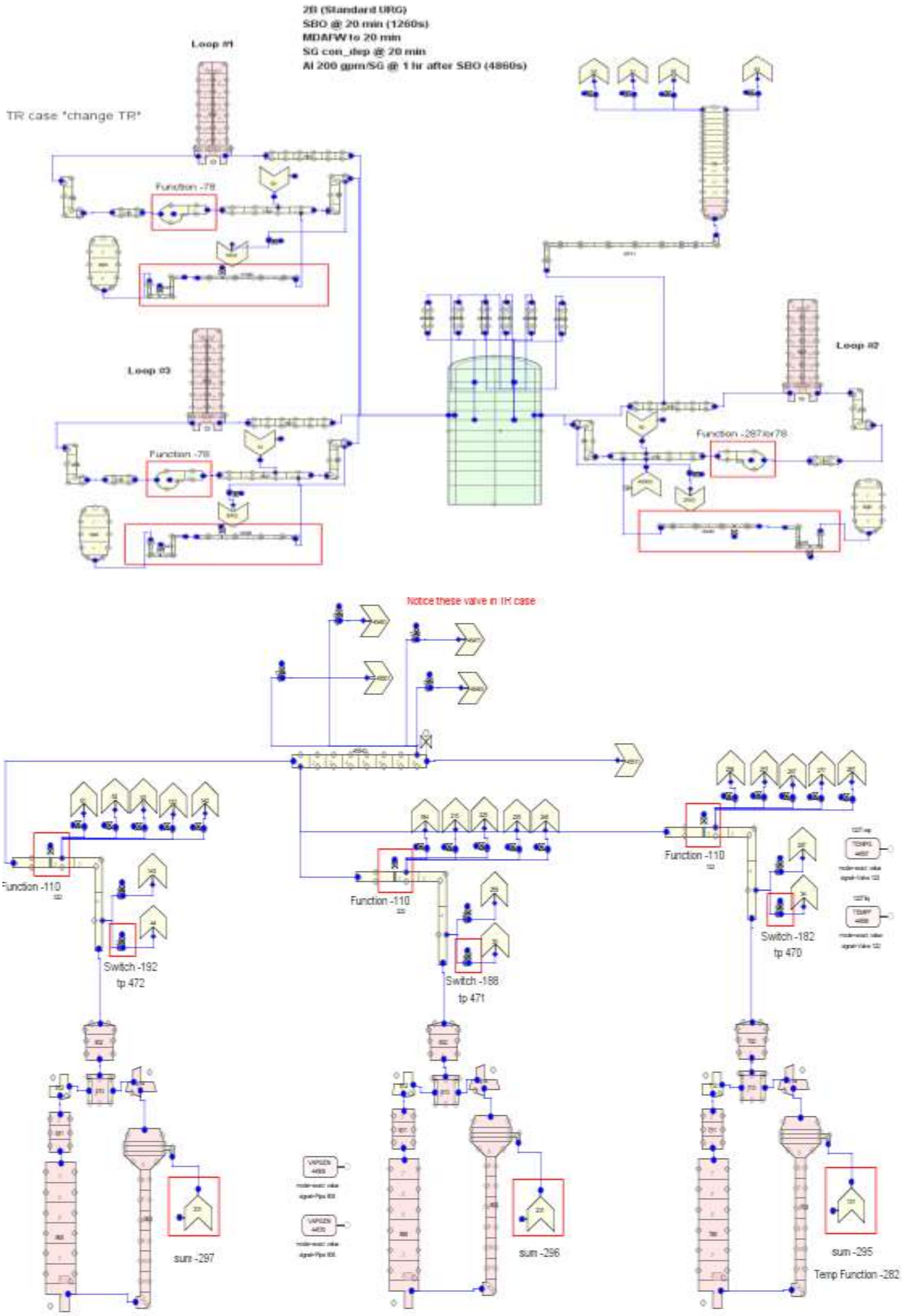

Figure 1   Schematic Diagram of Maanshan TRACE/SNAP Model

2.3 The Modeling of Maanshan SBO Event

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. 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 MWt) during the simulation.

2.4 The TRACE Results for the SBO Event

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. Due to TDAFW system, 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 2.

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 3. 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 4 show the cold leg liquid temperature for three RCS loops respectively. Figure 5 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. 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.

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, the TRACE results for SBO accident shows good agreement with the plant data. In addition, the results which are obtained by TRACE V5.0p3 / SNAP V2.2.1 are also shown in the Figure 2~5.

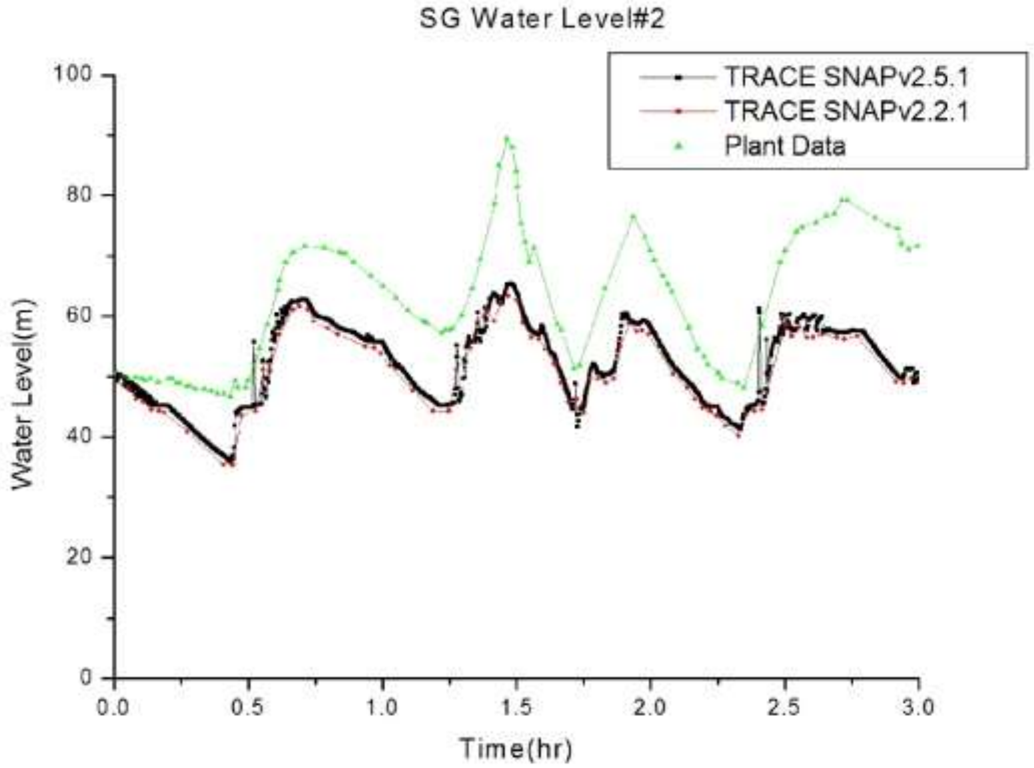

Figure 2 Steam Generator #2 Narrow Range Water Level

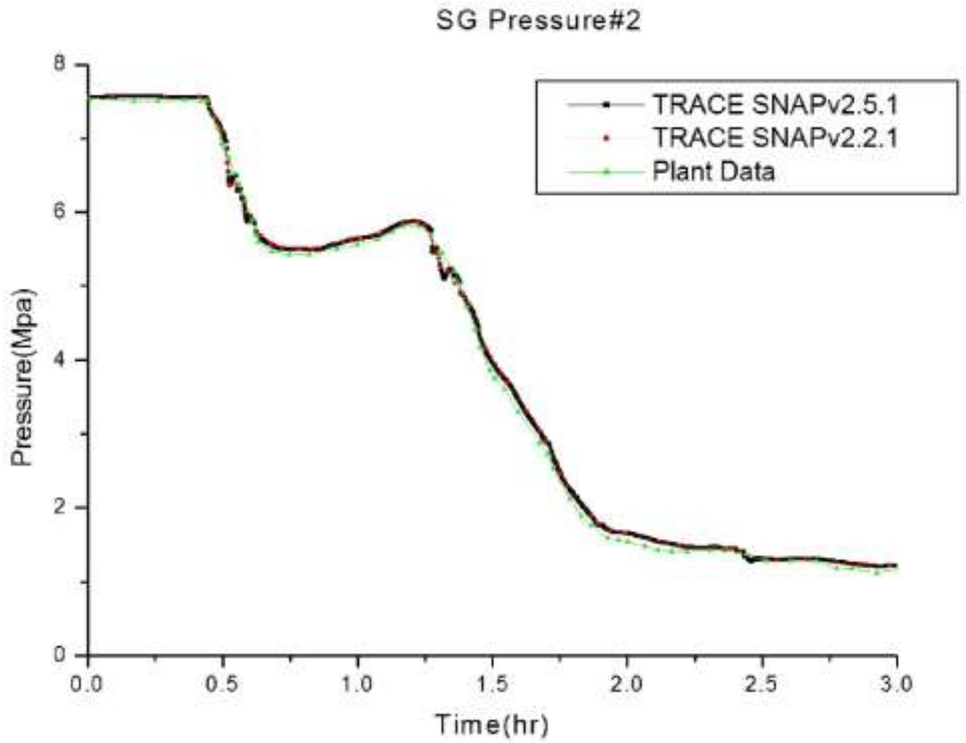

Figure 3 Steam Generator #2 Pressure

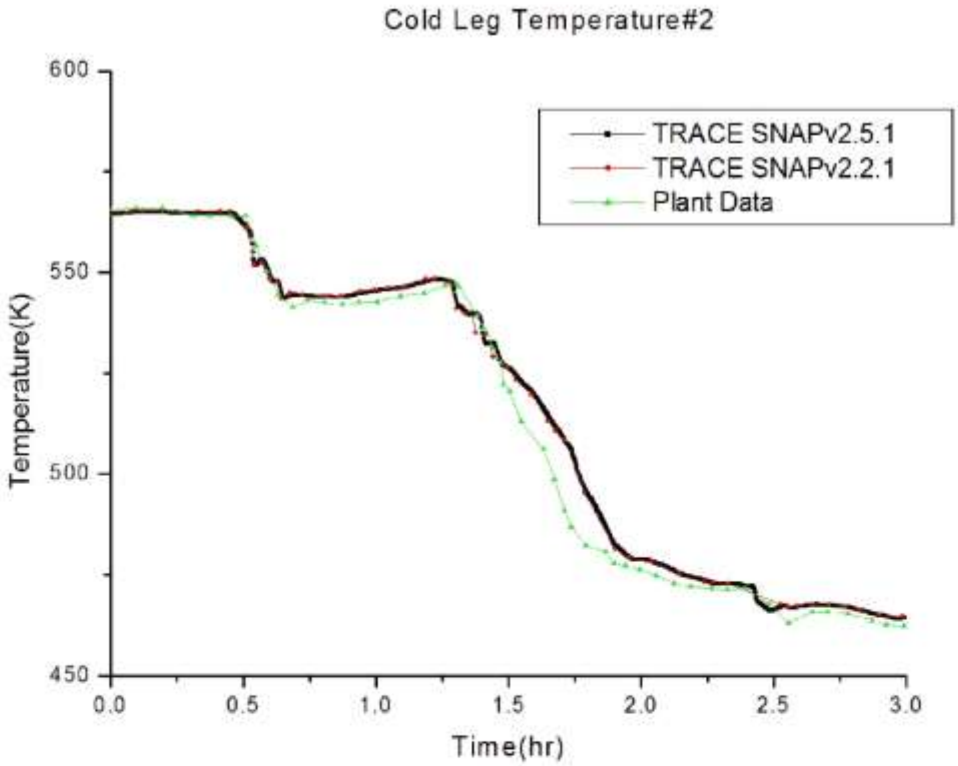

Figure 4 Cold Leg #2 Liquid Temperature

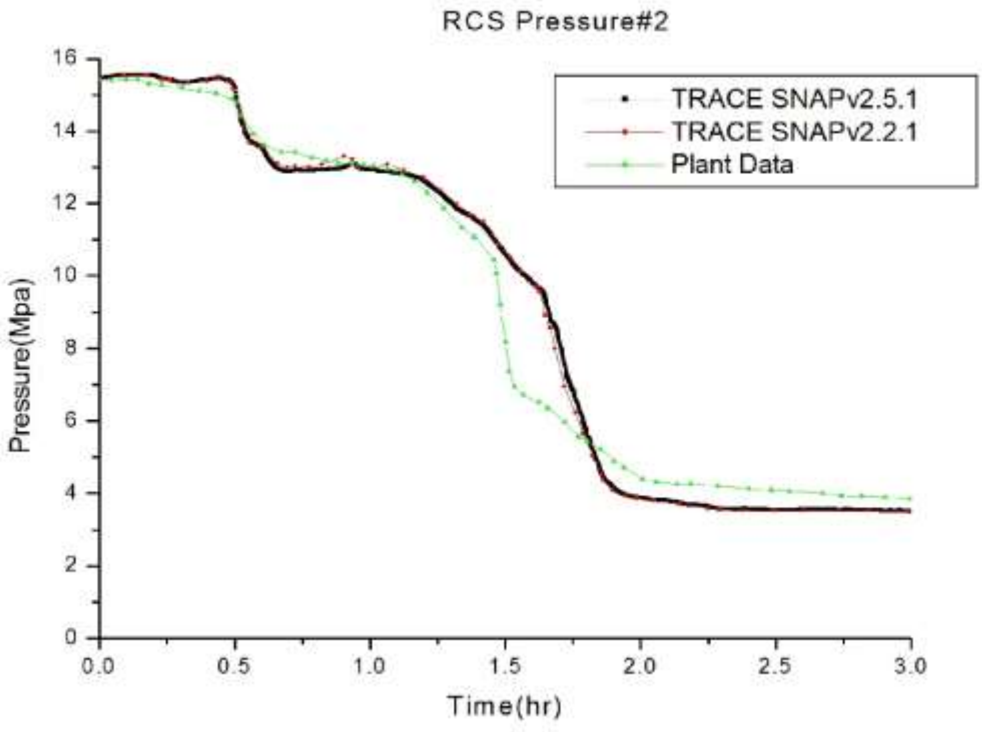

Figure 5 Reactor Coolant System Pressure

3 DESCRIPTION OF MODELING METHOD FOR ELAP EVENT WITH URG AND FLEX

To determine the severity of ELAP event and the mitigation capability of URG and FLEX strategy, this research will run a base case without any mitigation strategy, and four cases with multiple mitigation strategy. All the cases are in a situation of ELAP event. The scenario and assumptions of ELAP event in this study were referred to the WCAP-17601-P report.

3.1 Base Case (no mitigation strategy, Case 0)

In base case (Case 0), we assumed an earthquake occurred at 1 minute which resulted in the reactor scram, MSIV closure, turbine trip, feedwater trip, and RCP seal leakage. Table 3 shows the sequence of the base case. Seal begins to leak at a rate of 5gpm per loop at the moment and rises up to 21gpm per loop after 14 minute due to the flashing across seal face. After two hours, there is a control depressurization activated on the secondary side to maintain SG pressure at about 300 psia. In this case, the TDAFP and accumulator are assumed to be available at all times.

3.2 Mitigation Strategy with URG or FLEX (Case 1~4)

The main difference between URG and FLEX is that FLEX owing to the FSG (FLEX Support Guideline) high pressure injection equipment, there is no need to perform emergency depressurization. The URG was simulated in Case 1 and 2. The FLEX was simulated in Case 3 and 4. The sequence of each case is listed on Table 4.

In cases with multiple mitigation strategy, unlike base case, we assume that seal begins to leak at a rate of 21gpm per loop as earthquake occurred at 1 minute. Control depressurization will also be activated at the moment. In Case 1 and 2, each case will follow URG strategy while the ELAP event lasts for more than 8 and 24 hours. Otherwise, in Case 3 and 4, we assume that the plant has FSG high pressure injection equipment. Each case will follow FLEX strategy once ELAP event lasts for more than 8 and 24 hours.

Table 3 The Sequence of Base Case

Event (Base case)	Time(min)
Start of simulation	0
The SBO, reactor scram, turbine trip, MSIV closure, RCP trip, feedwater trip, and seal leakage occur. Seal leakage rate is 5 gpm/loop	1
Seal leakage rate rise up to 21 gpm/loop	14
Control depressurization, SG pressure depressurize to 21 kg/cm^2 (300 psia)	120
End of simulation	…

*The TDAFP and accumulator is available.

Table 4 The Sequences of Case 1~4

Event	Time (min)			
	Case 1	Case 2	Case 3	Case 4
Start of simulation	0			
The SBO, reactor scram, turbine trip, MSIV closure, RCP trip, feedwater trip, and seal leakage occur TDAFP on Seal leakage rate is 21 gpm/loop Control depressurization, SG pressure depressurize to 21 kg/cm² (300 psia)	1			
Emergency depressurization, SG pressure depressurize to 3 kg/cm²	480(8hr)	1440(24hr)	x	x
TDAFP off Fire pump 800 gpm (35.704 Kg/s) to SG Hydro-Test pump 25 gpm (1.14 Kg/s) to RPV	480(8hr)	1440(24hr)	x	x
TDAFP off FSG pump 215 gpm(9.595 Kg/s) to SG FSG pump 40 gpm (1.79 Kg/s) to RPV	x	x	480(8hr)	1440(24hr)
End of simulation	4800(80hr)			

4 RESULTS

4.1 The Results of Case 0

Figure 6~11 present the TRACE analysis results for Case 0. In this case, the SBO occurred at 1 minute. Then, the reactor scram and control depressurization occurred which caused the SG and RCS pressures to decrease (see Fig. 6). The SG water level kept at full water level since TDAFP was available (see Fig. 7). In base case, without any mitigation measure, RCS water level dropped down below TAF at 61.89hr because of the seal leakage (see Fig. 8). It should be noticed that once RCS water level dropped down to the elevation of seal break (7.92m), there would be an oscillation on seal leakage rate since there were no coolant but steam still leak out from seal at the moment (shown in Fig. 9). Fig. 10 shows the water level of the accumulator. The accumulator started to inject water at about 3.6 hr. The PCT started to increase after the RCS water level lower than TAF (shown in Fig. 11).

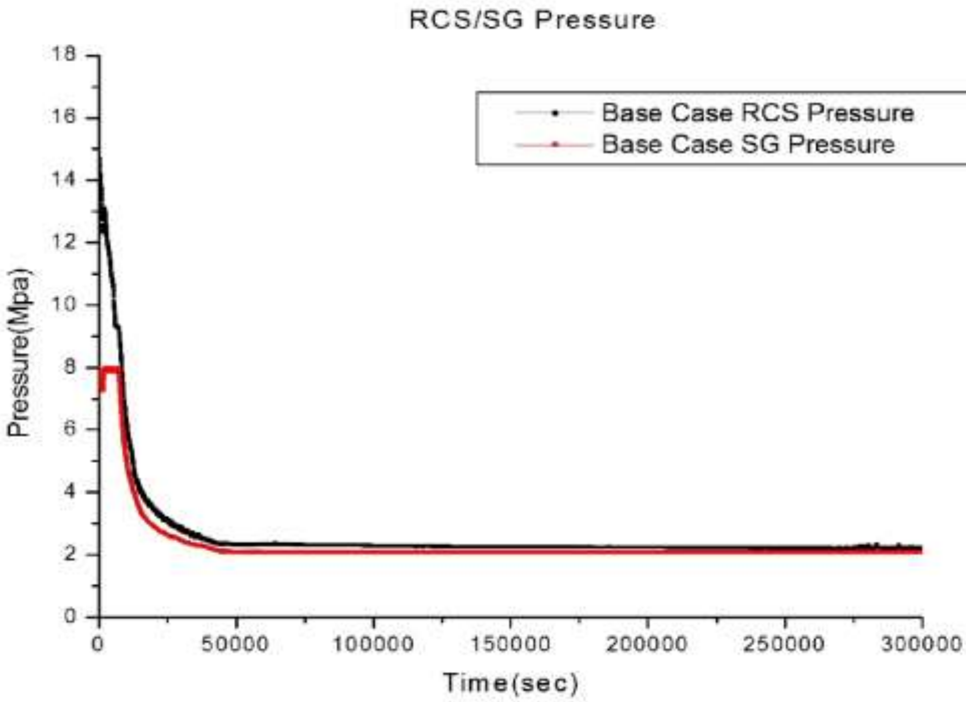

Figure 6 RCS and SG Pressure of Case 0

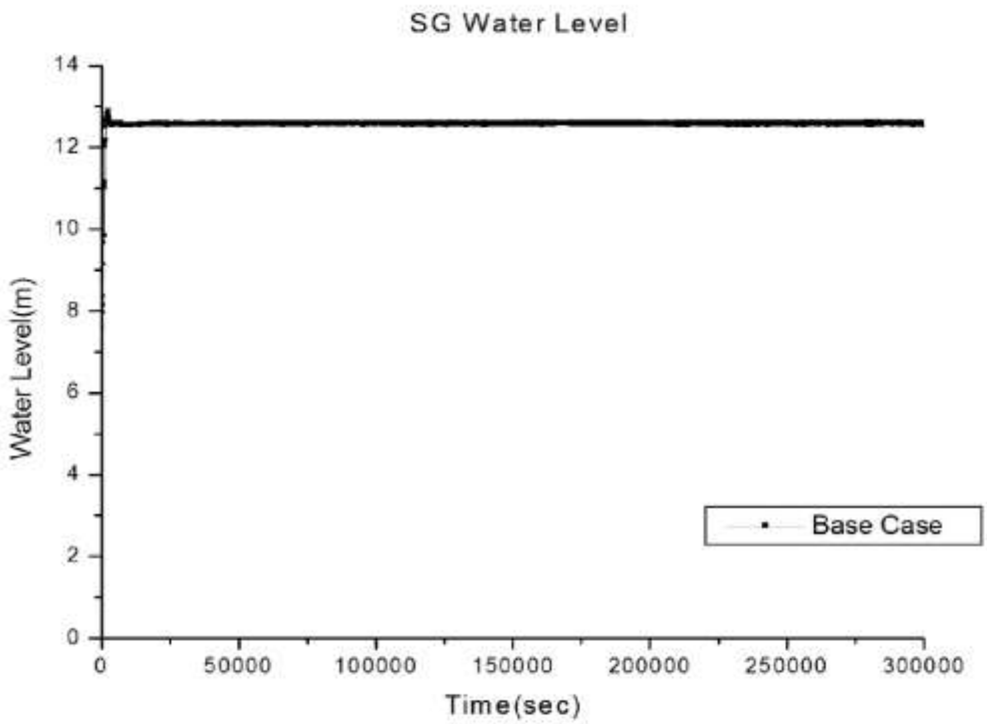

Figure 7 SG Water Level of Case 0

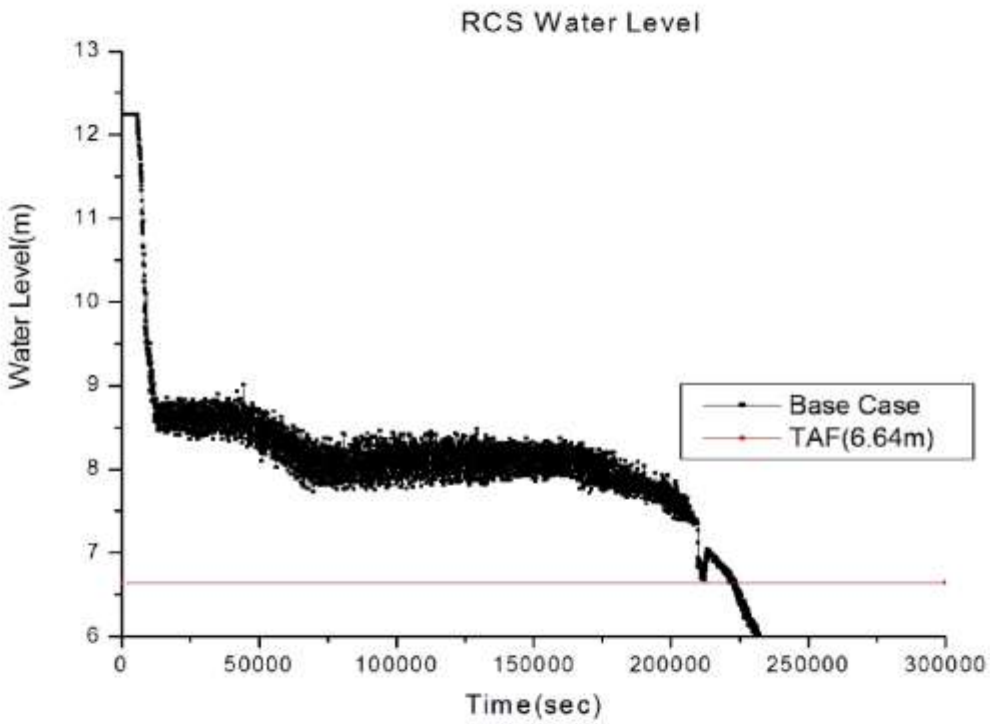

Figure 8 RCS Water Level of Case 0

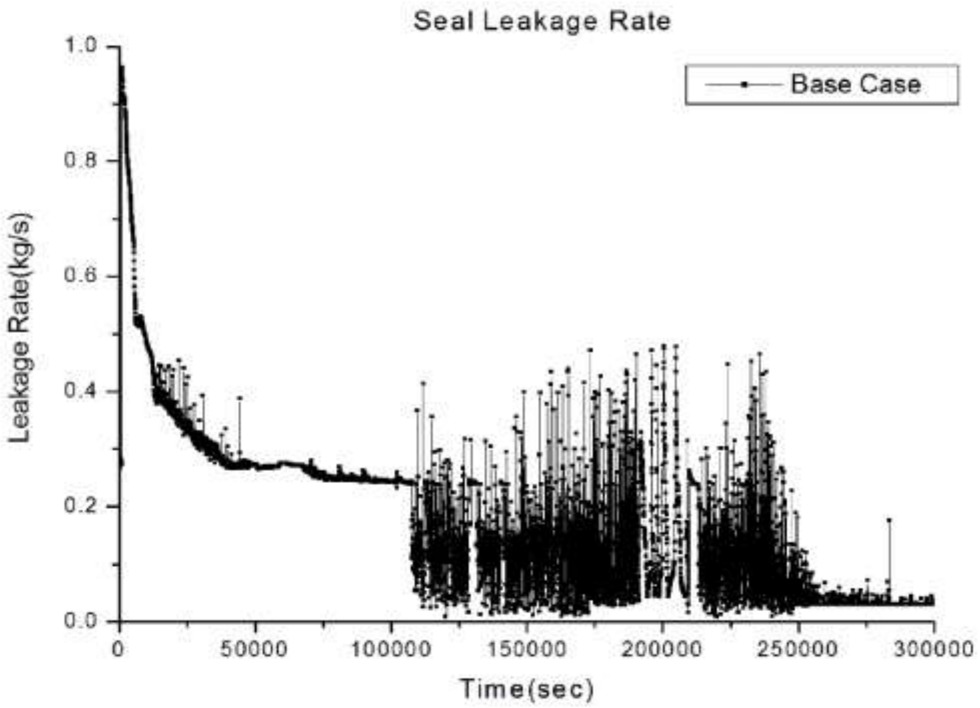

Figure 9   Seal Leakage Rate of Case 0

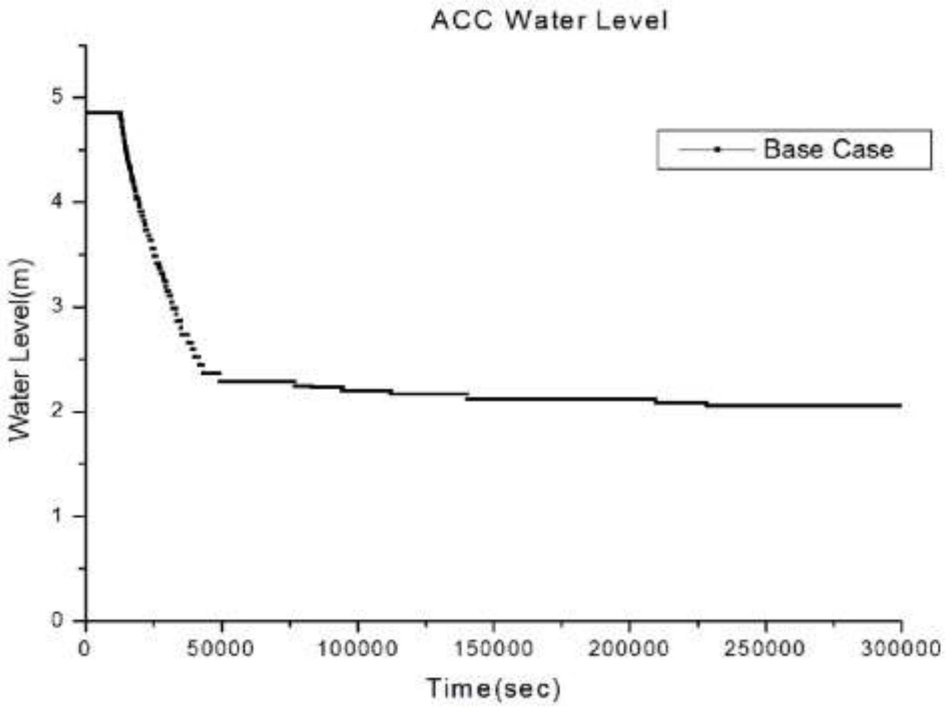

Figure 10 ACC Water Level of Case 0

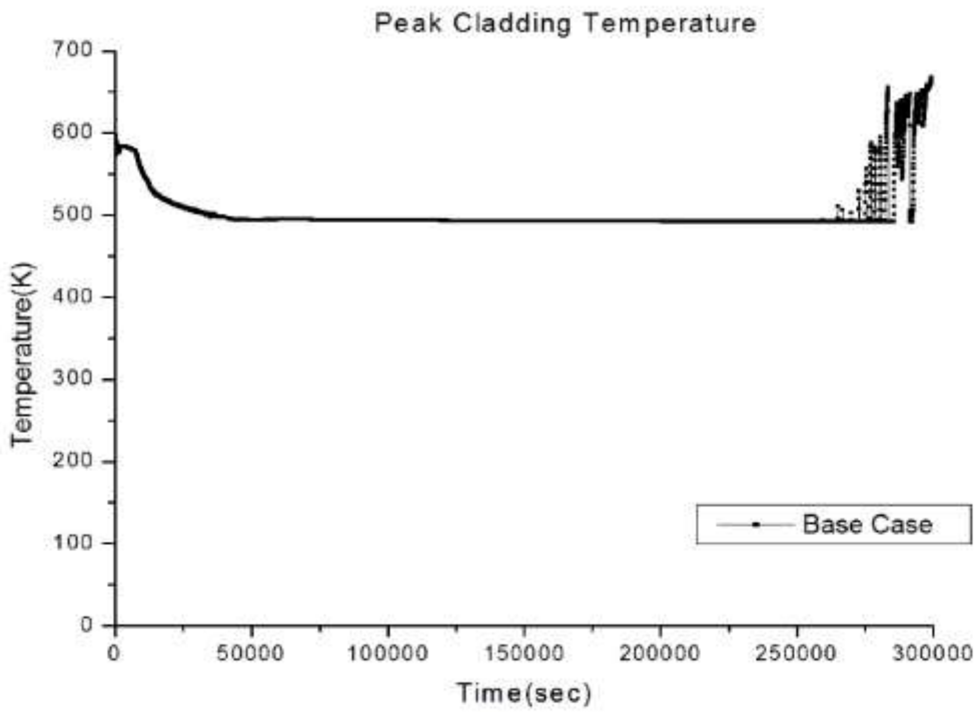

Figure 11 Peak Cladding Temperature of Case 0

4.2 <u>The Results of Case 1</u>

Figure 12~17 illustrate the TRACE analysis results for Case 1. The URG was simulated in Case 1. Therefore, the control and emergency depressurizations were performed in this case. In addition, the water injection of fire and Hydro-Test pumps (at 8 hr, see Table 4) were performed in this case. In this case, the SBO occurred at 1 minute first. Subsequently, the reactor scram and control depressurization occurred which resulted in the RCS and SG pressures dropping (see Fig. 12). The RCS and SG pressures decreased again due to the emergency depressurization after 8 hr. The SG water level kept at full water level since TDAFP was available (see Fig. 13) during 0 minute~8 hr. After 8hr, the fire pump injected water to SGs and Hydro-Test pump injected water to RPV. Therefore, the SG water level still kept at full water level and RCS water level increased after 8 hr (shown in Fig. 13 and 14). The seal leakage rate result is shown in Fig. 15. The RCS pressure affects the seal leakage rate. Fig. 16 shows the water level of the accumulator. The accumulator started to inject water at about 0.2 hr. The PCT is always under 1088.7 K because the RCS water level is above the TAF (see Fig. 17).

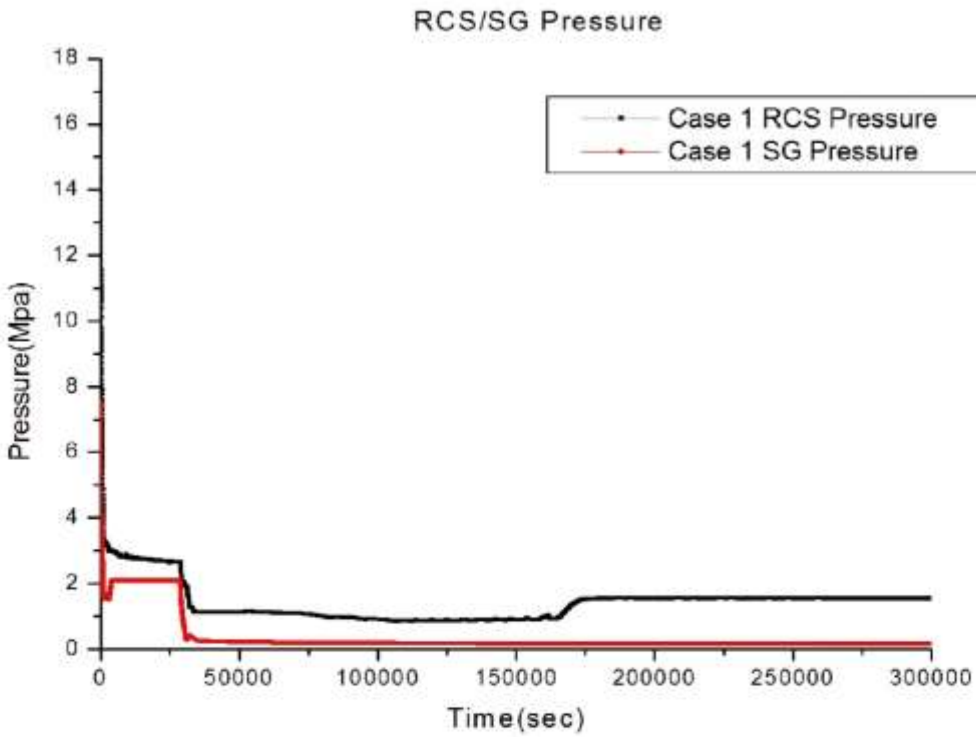

Figure 12 RCS and SG Pressure of Case 1

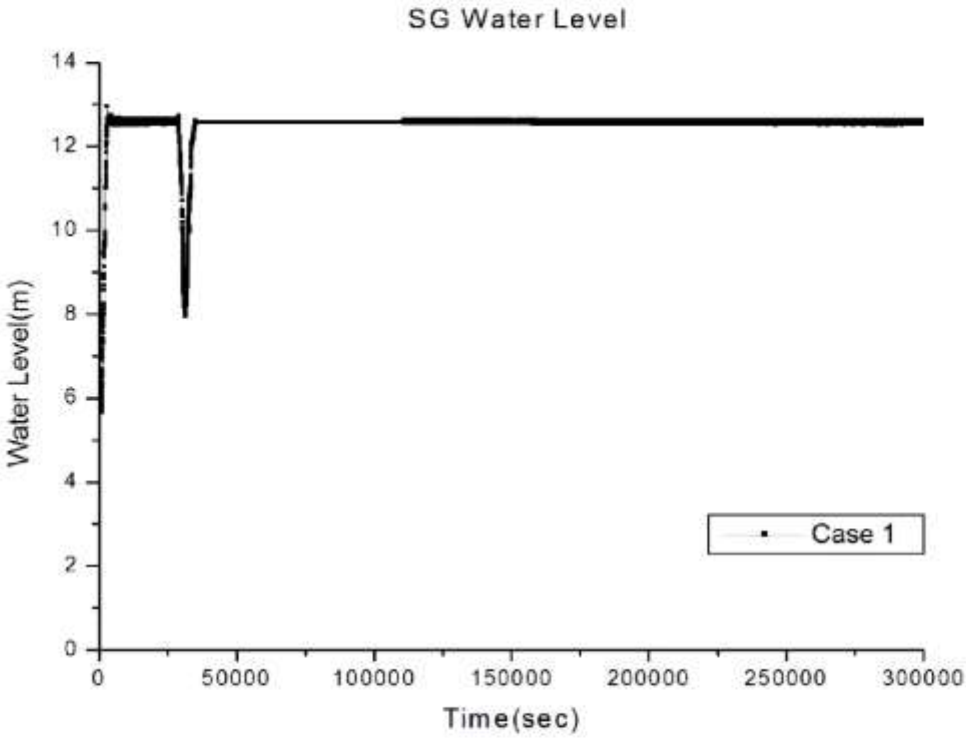

Figure 13 SG Water Level of Case 1

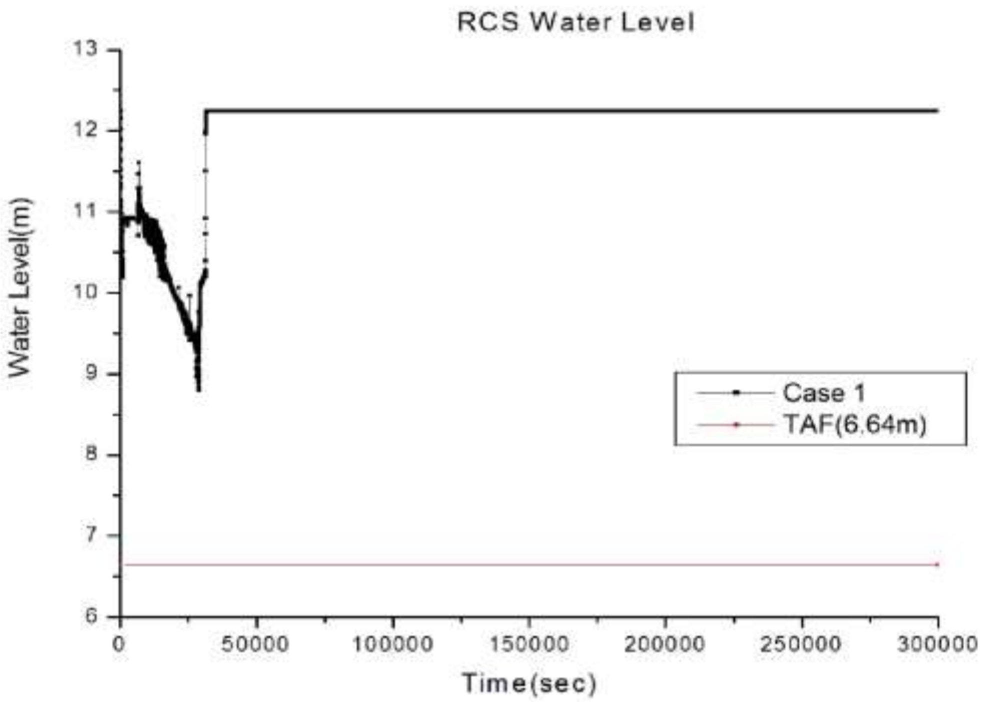

Figure 14 RCS Water Level of Case 1

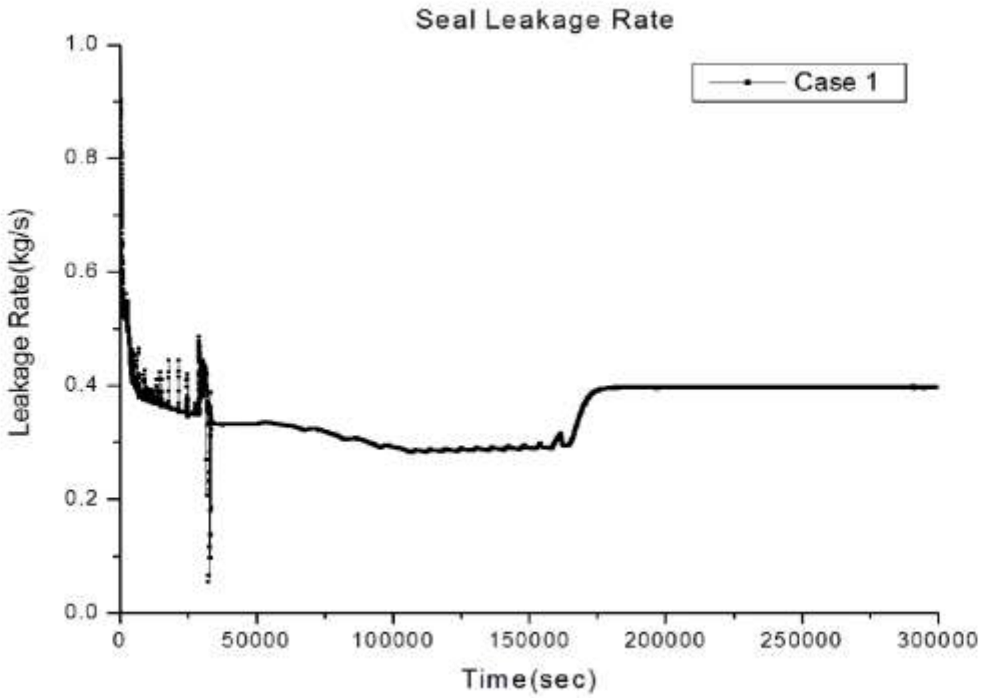

Figure 15 Seal Leakage Rate of Case 1

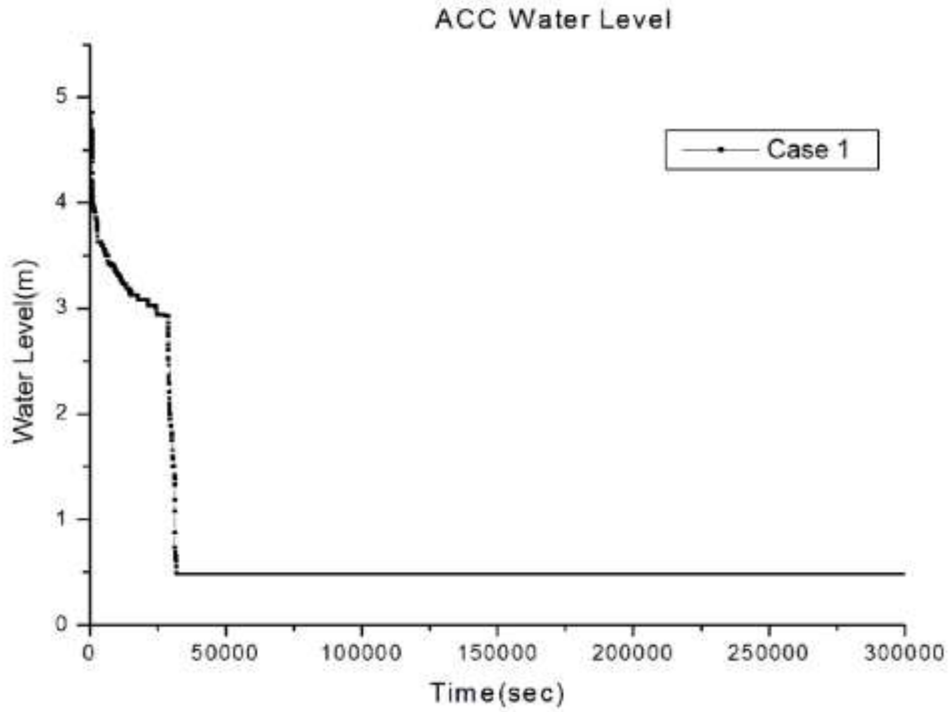

Figure 16 ACC Water Level of Case 1

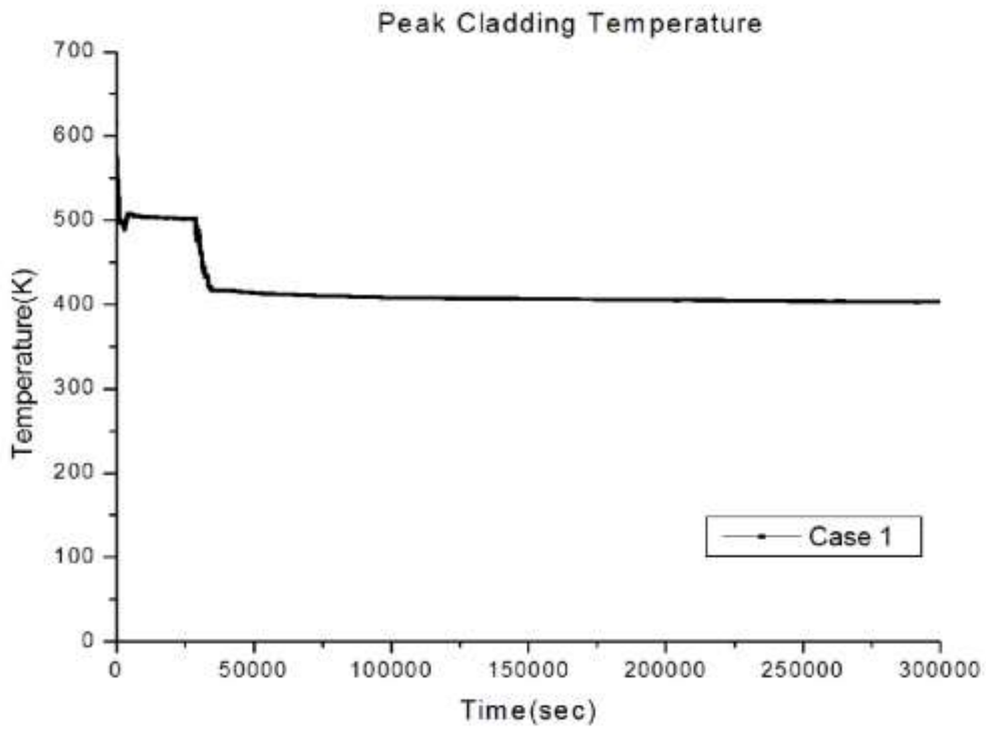

Figure 17 Peak Cladding Temperature of Case 1

4.3 The Results of Case 2

Figure 18~23 show the TRACE analysis results for Case 2. The URG was simulated in Case 2. Hence, the control and emergency depressurizations were performed in this case. In addition, the water injection of fire and Hydro-Test pumps (at 24 hr, see Table 4) were performed in this case. In this case, the SBO occurred at 1 minute first. Then, the reactor scram and control depressurization occurred which resulted in the RCS and SG pressures decrease (see Fig. 18). The RCS and SG pressures went down again due to the emergency depressurization after 24 hr. The SG water level kept at full water level since TDAFP was available (see Fig. 19) during 0 minute~24 hr. After 24hr, the fire pump injected water to SGs and Hydro-Test pump injected water to RPV. Therefore, the SG water level still kept at full water level and RCS water level increased after 24 hr (shown in Fig. 19 and 20). The seal leakage rate result is shown in Fig. 21. The RCS pressure affects the seal leakage rate. Fig. 22 presents the water level of the accumulator. The accumulator began to inject water at about 0.2 hr. The PCT is always under 1088.7 K because the RCS water level is above the TAF (see Fig. 23).

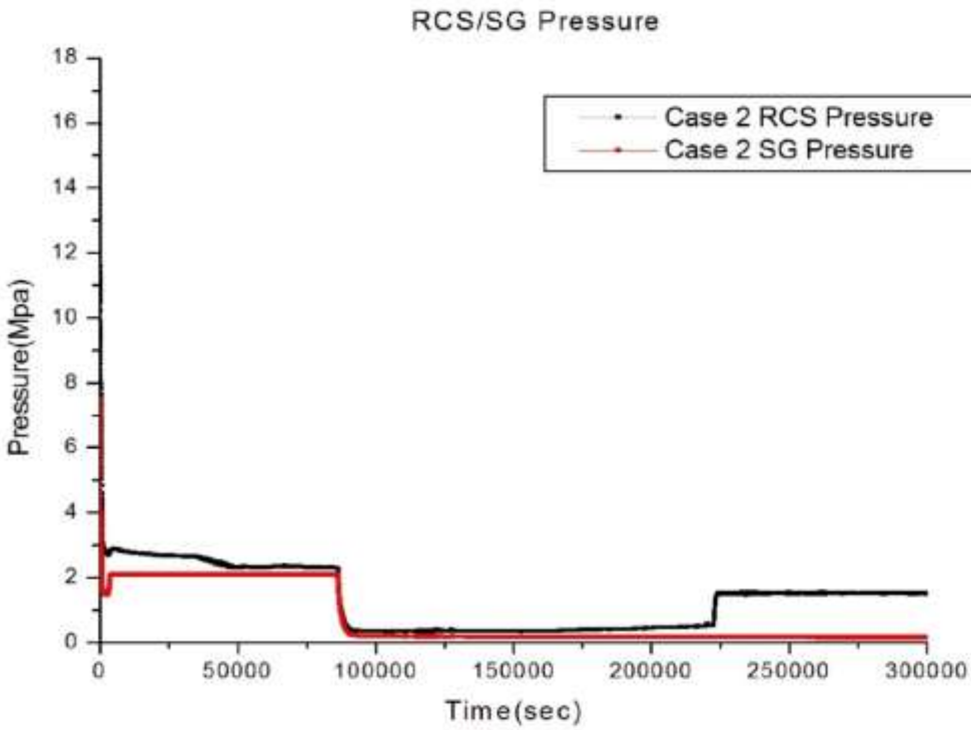

Figure 18 RCS and SG Pressure of Case 2

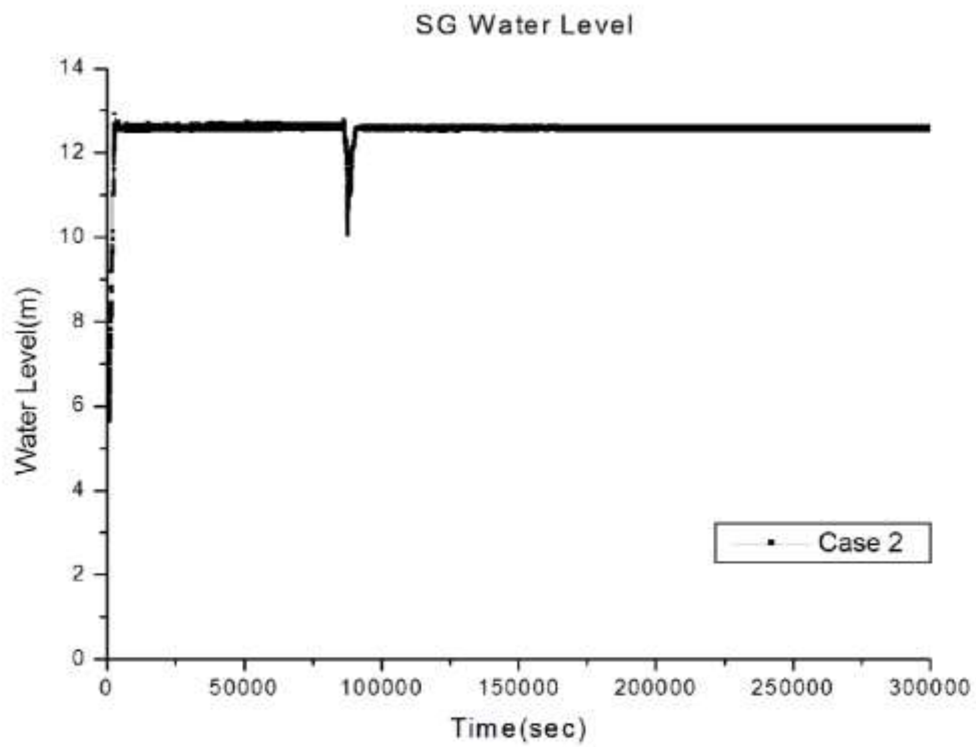

Figure 19 SG Water Level of Case 2

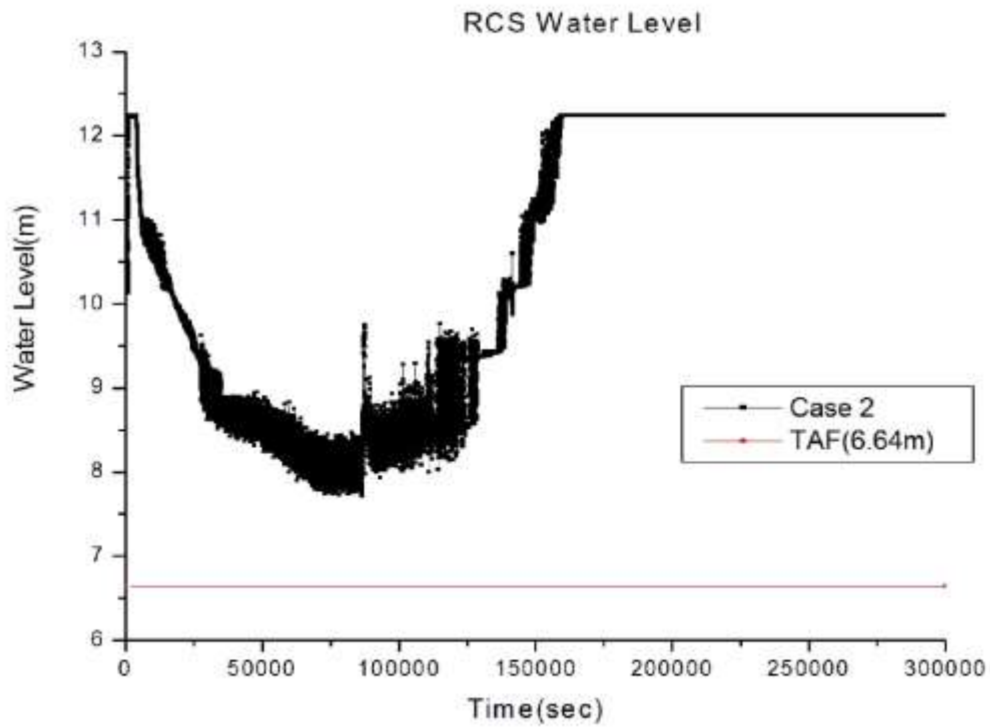

Figure 20 RCS Water Level of Case 2

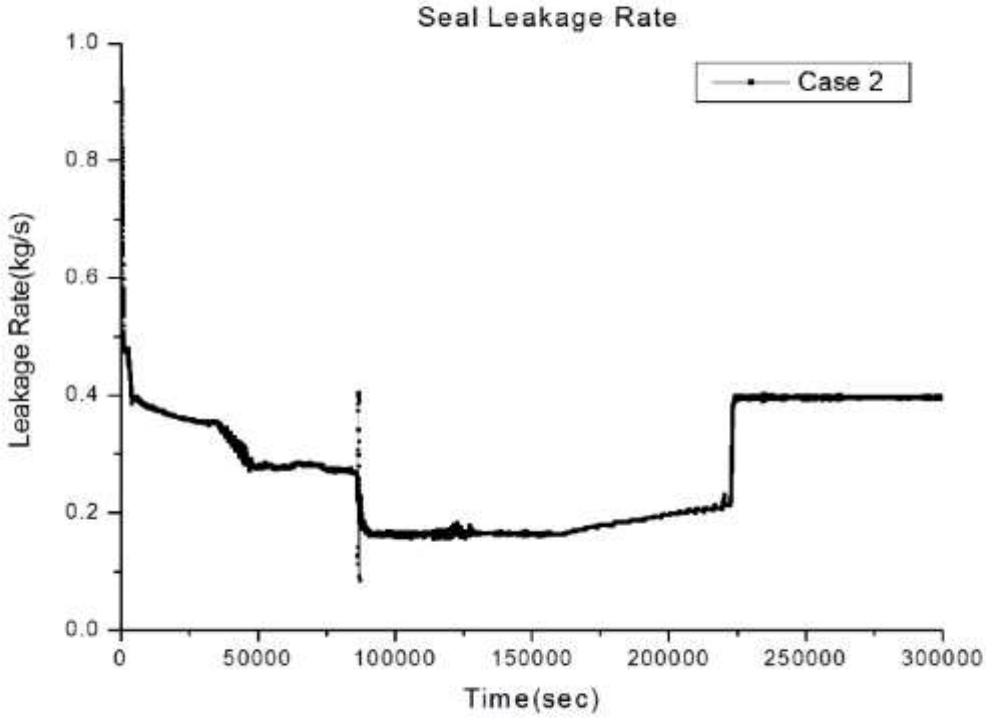

Figure 21 Seal Leakage Rate of Case 2

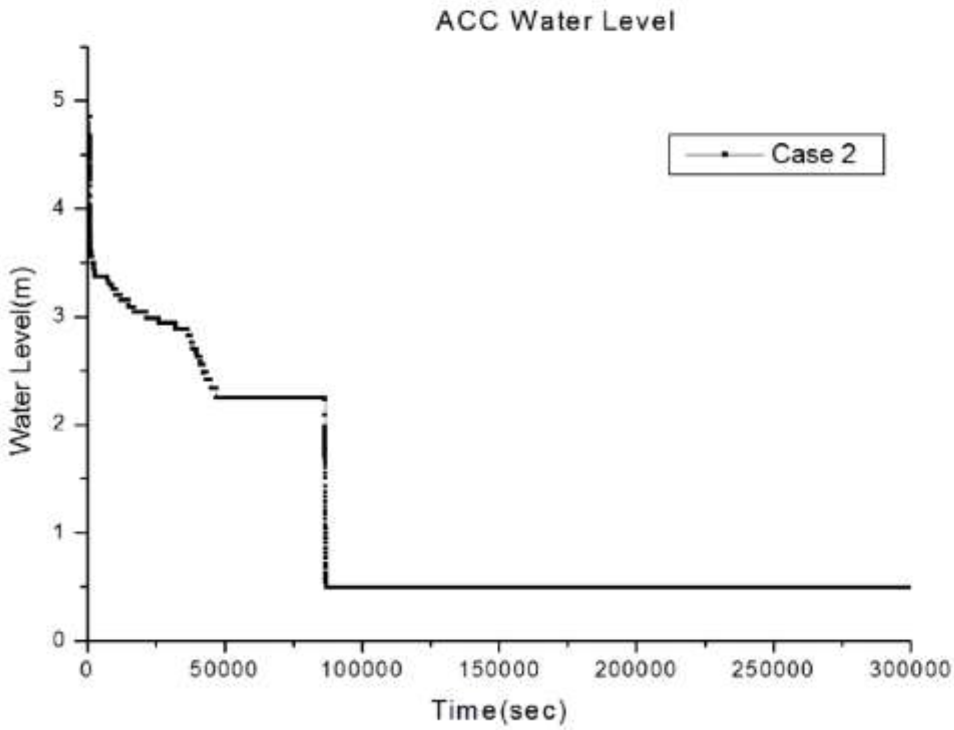

Figure 22 ACC Water Level of Case 2

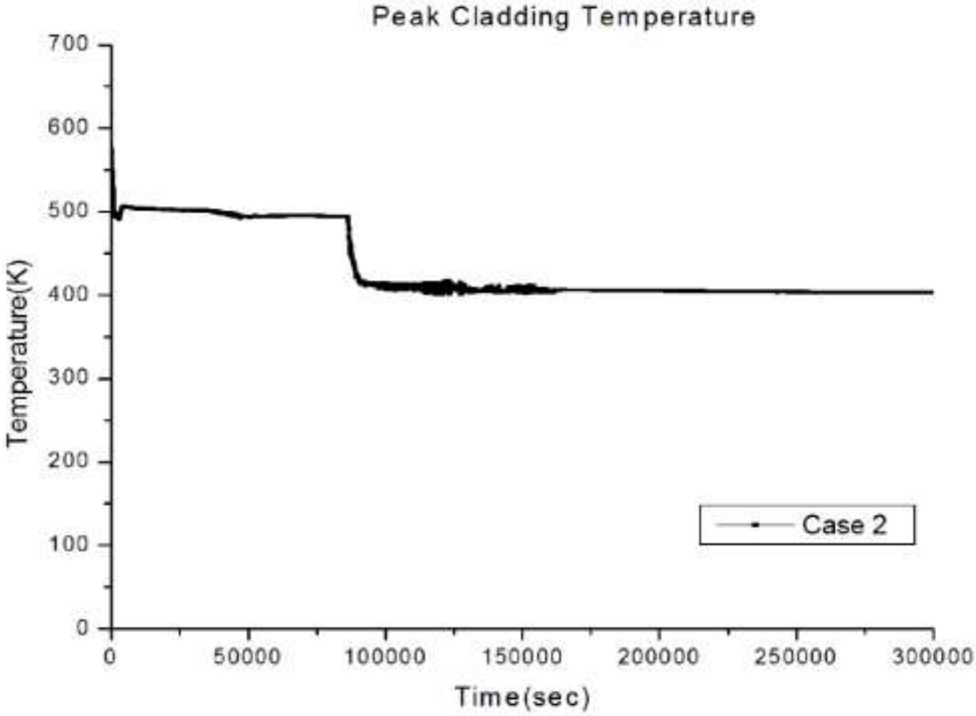

Figure 23 Peak Cladding Temperature of Case 2

4.4 <u>The Results of Case 3</u>

Figure 24~29 present the TRACE analysis results for Case 3. The FLEX was simulated in this case. Therefore, the control depressurization was performed in this case. In addition, the water injection of FSG pumps (at 8 hr, see Table 4) were also performed in this case. In this case, the SBO occurred at 1 minute first. Subsequently, the reactor scram and control depressurization occurred which caused the RCS and SG pressures to drop (see Fig. 24). The SG water level kept at full water level since TDAFP was available (see Fig. 25) during 0 minute~8 hr. After 8hr, the FSG pump (215 gpm) injected water to SGs and FSG pump (40 gpm) injected water to RPV. Therefore, the SG water level still kept at full water level and RCS water level increased after 8 hr (shown in Fig. 25 and 26). Fig. 27 shows the seal leakage rate result. The RCS pressure affects the seal leakage rate. Fig. 28 illustrates the water level of the accumulator. The accumulator started to inject water at about 0.2 hr. The PCT is always under 1088.7 K because the RCS water level is above the TAF (see Fig. 29).

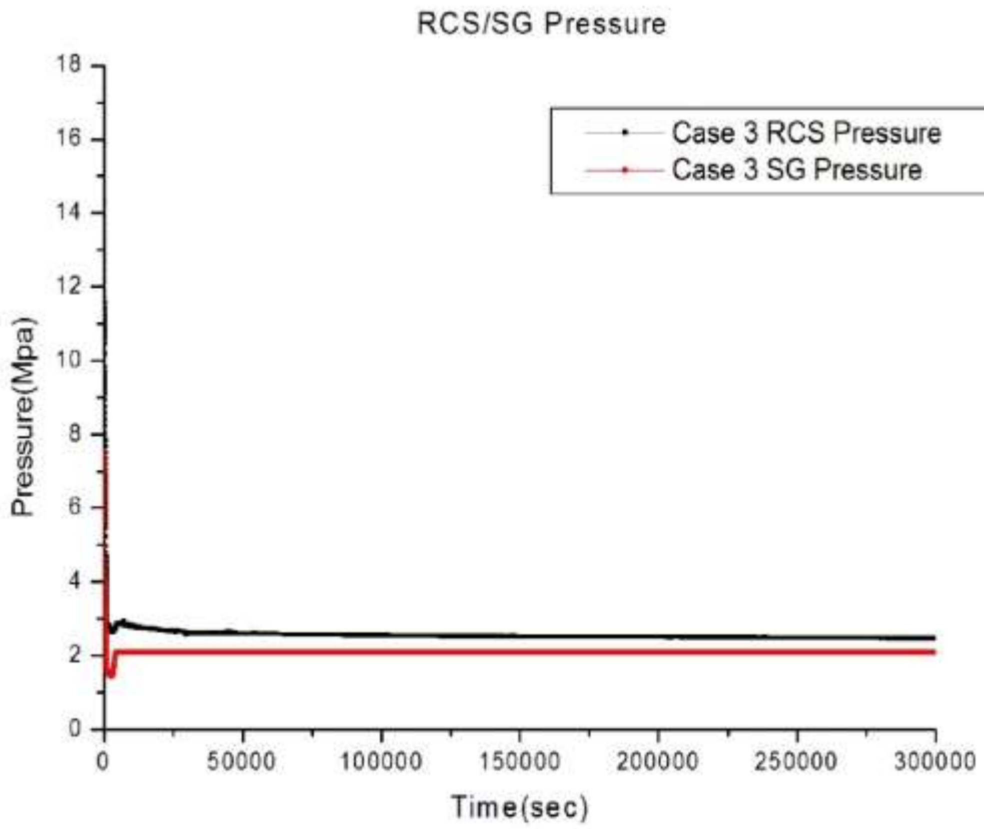

Figure 24 RCS and SG Pressure of Case 3

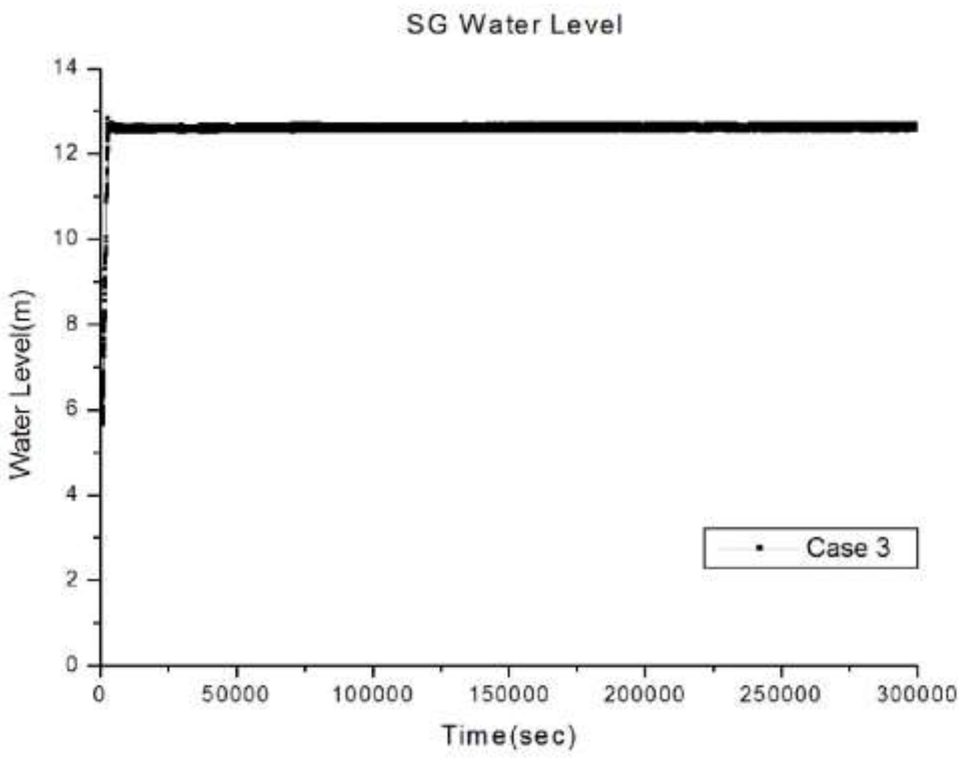

Figure 25 SG Water Level of Case 3

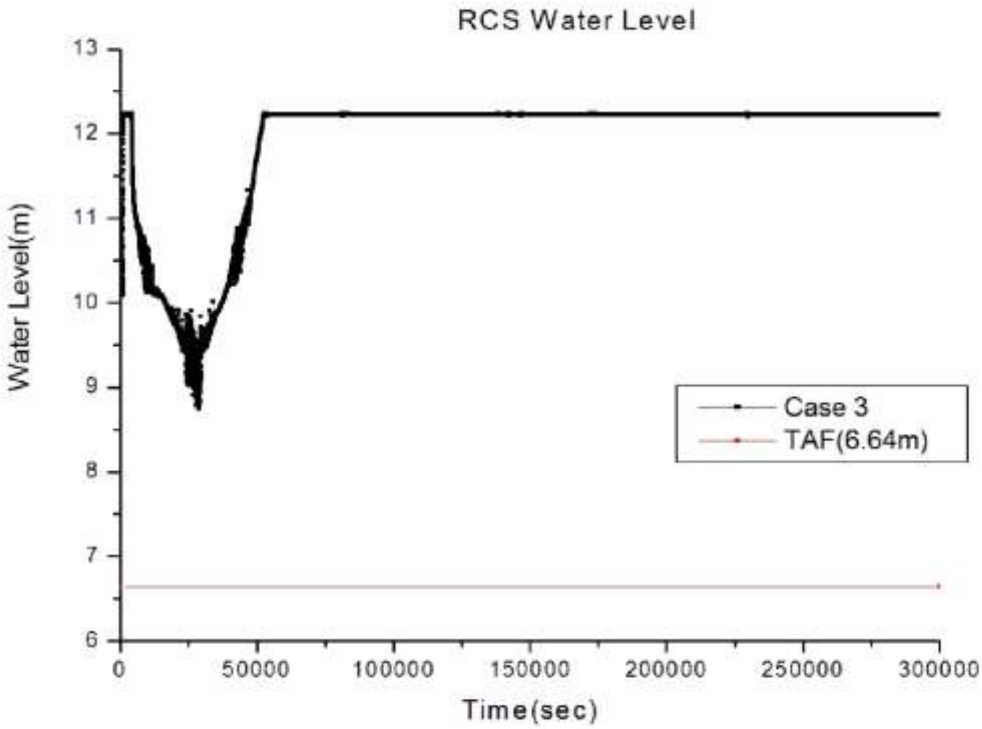

Figure 26 RCS Water Level of Case 3

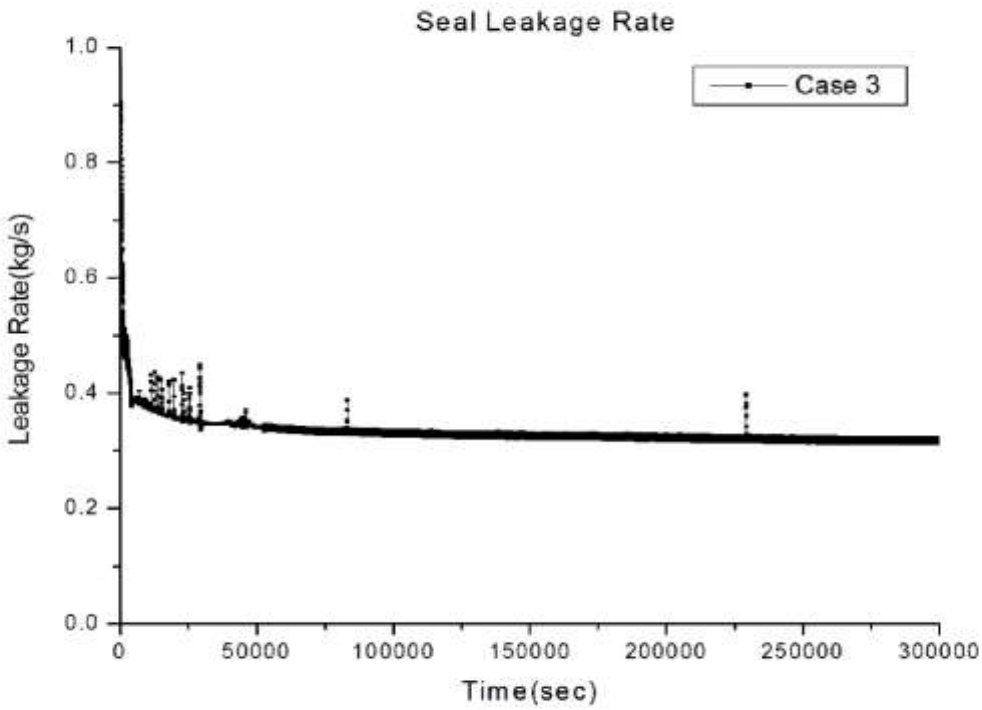

Figure 27 Seal Leakage Rate of Case 3

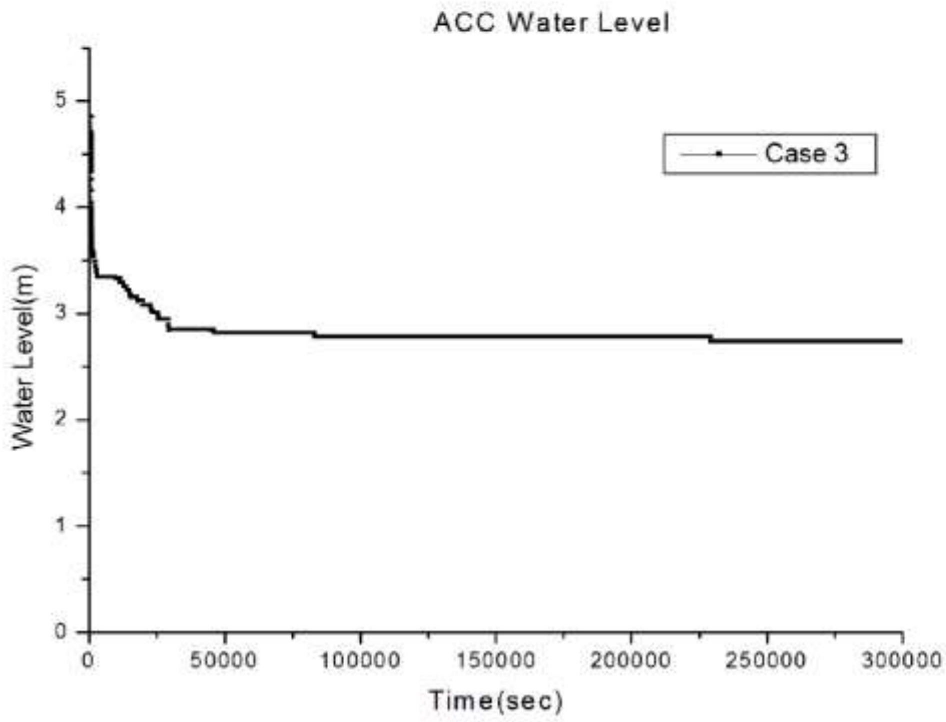

Figure 28 ACC Water Level of Case 3

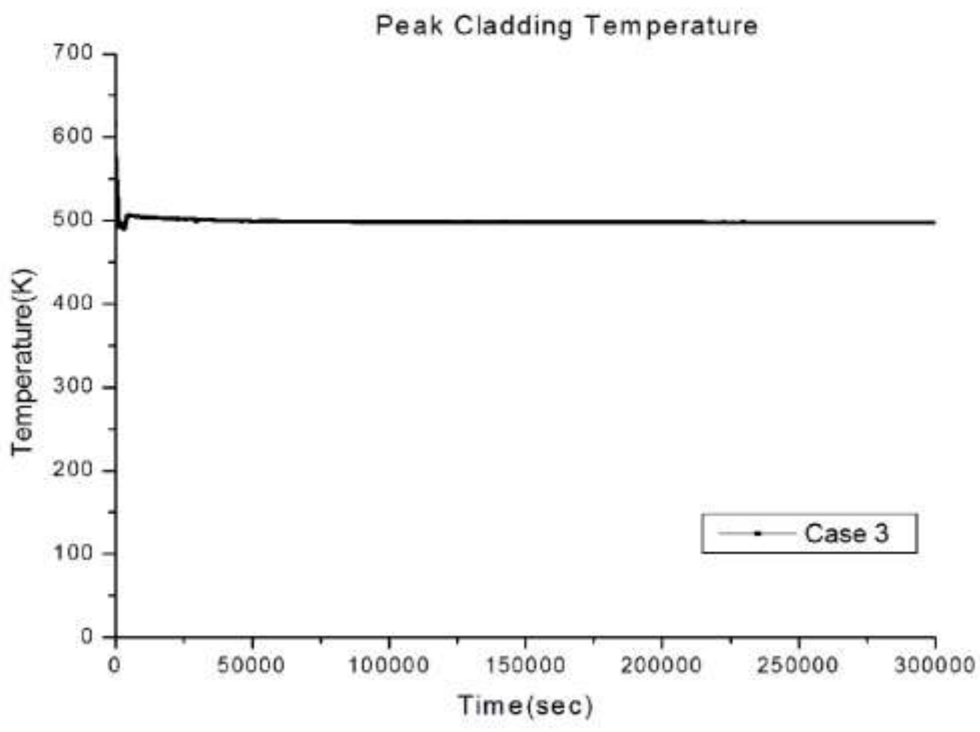

Figure 29 Peak Cladding Temperature of Case 3

4.5 The Results of Case 4

Figure 30~35 illustrate the TRACE analysis results for Case 4. The FLEX was simulated in Case 4. Therefore, the control depressurization was performed in this case. In addition, the FSG pumps (at 24 hr, see Table 4) were performed in this case. In this case, the SBO occurred at 1 minute first. Then, the reactor scram and control depressurization occurred which resulted in the RCS and SG pressures dropping (see Fig. 30). The SG water level kept at full water level since TDAFP was available (see Fig. 31) before 24 hr. After 24 hr, the FSG pump (215 gpm) injected water to SGs and FSG pump (40 gpm) injected water to RPV. Therefore, the SG water level still kept at full water level and RCS water level increased after 24 hr (see Fig. 31 and 32). The seal leakage rate result is shown in Fig. 33. Fig. 34 depicts the water level of the accumulator. The accumulator began to inject water at about 0.2 hr. The PCT is always lower than 1088.7 K because the RCS water level is above the TAF (see Fig. 35).

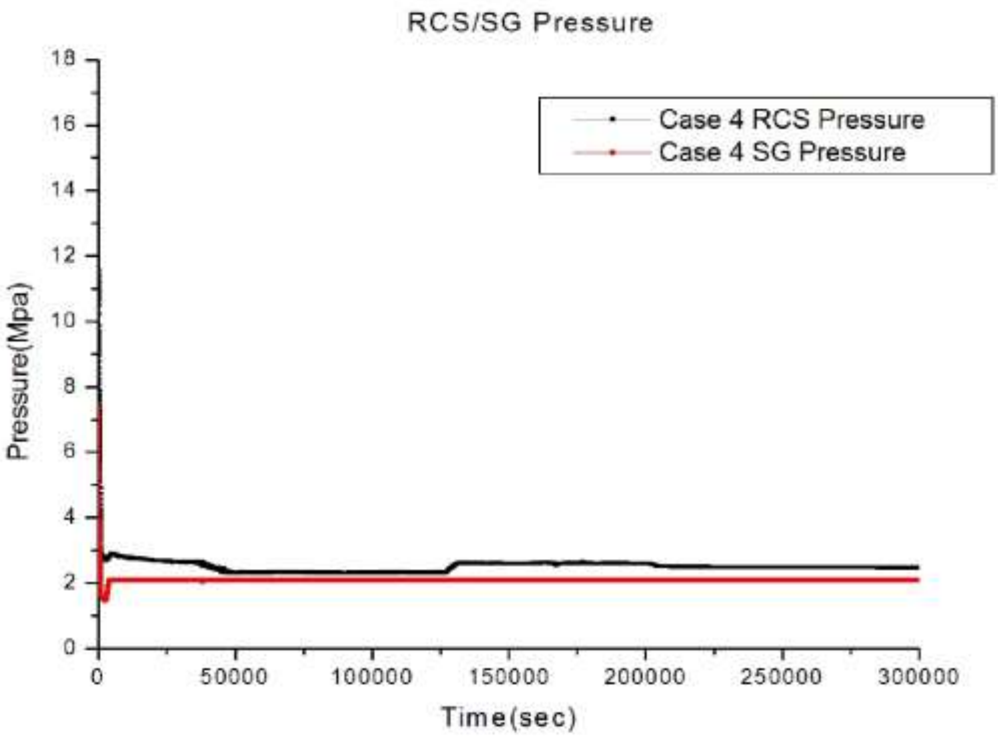

Figure 30 RCS and SG Pressure of Case 4

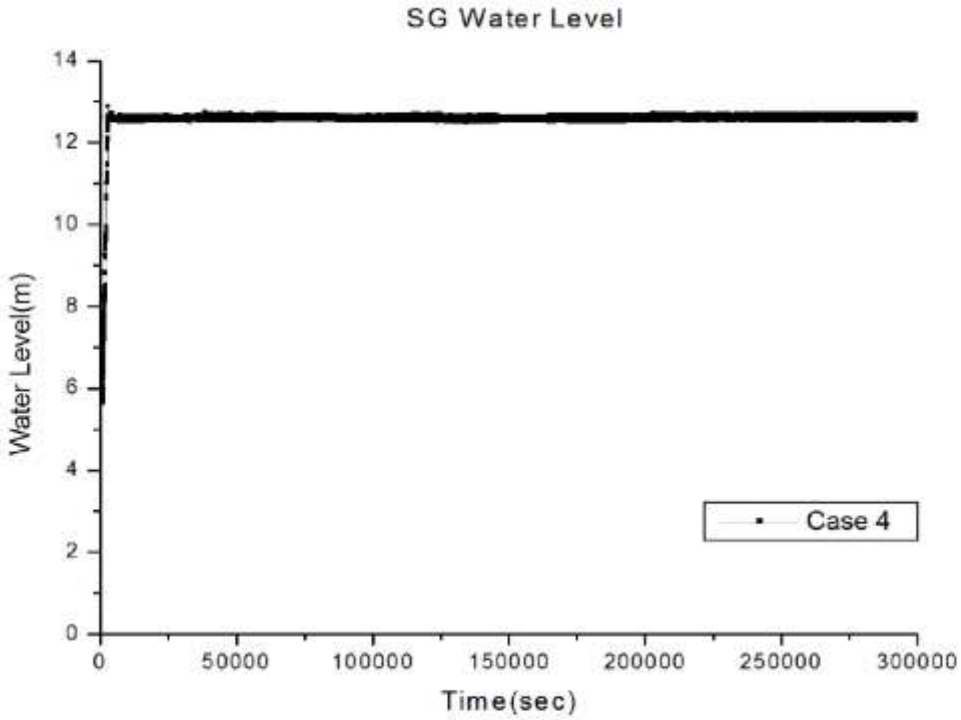

Figure 31 SG Water Level of Case 4

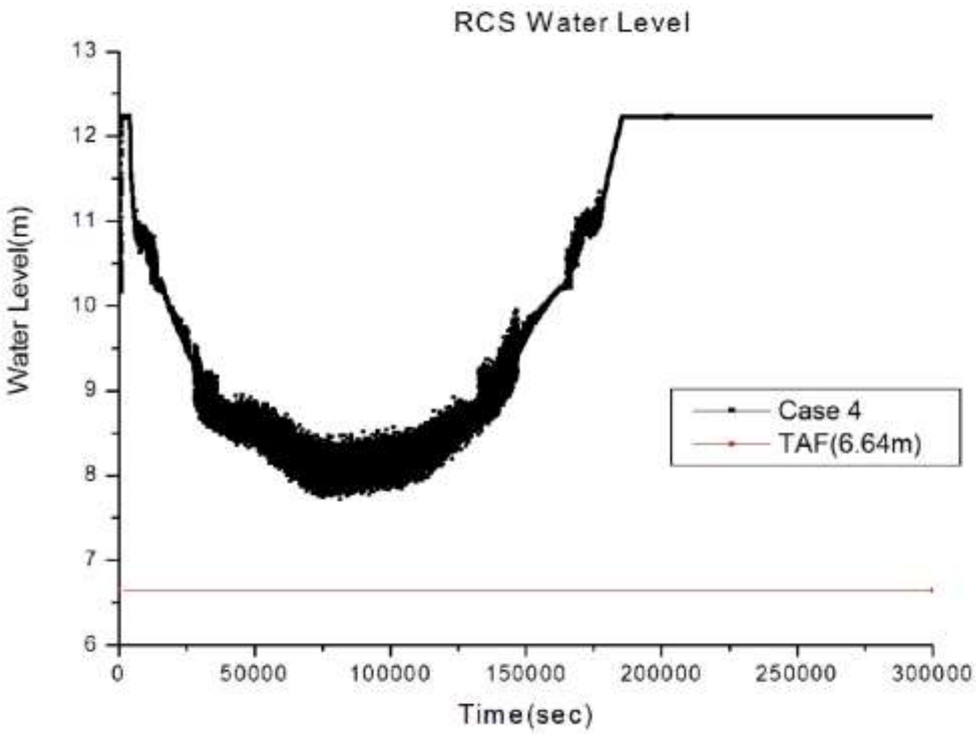

Figure 32 RCS Water Level of Case 4

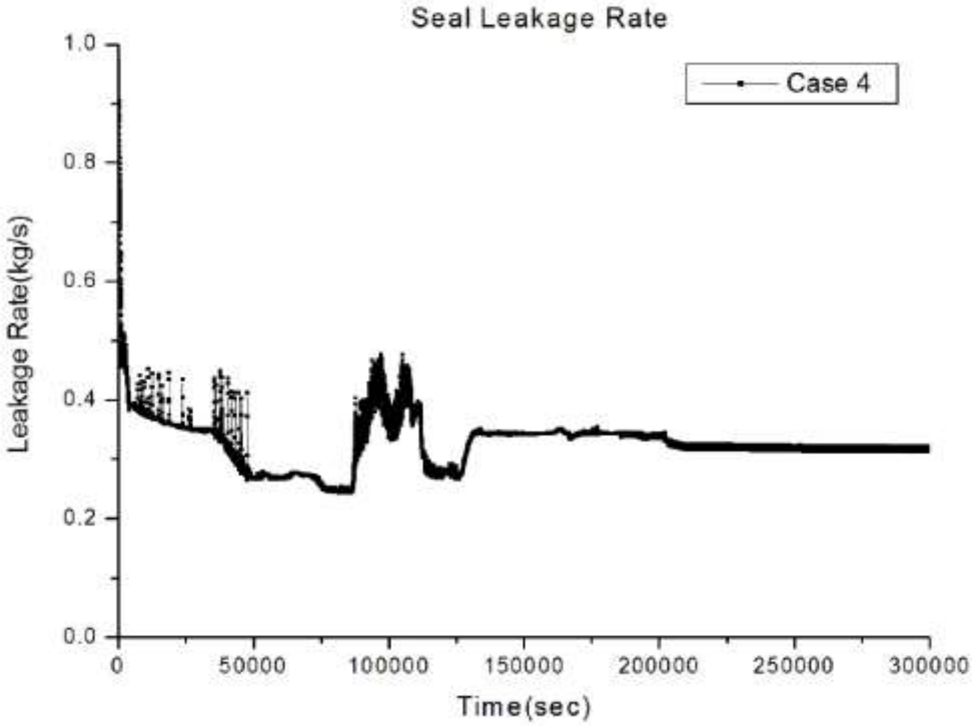

Figure 33 Seal Leakage Rate of Case 4

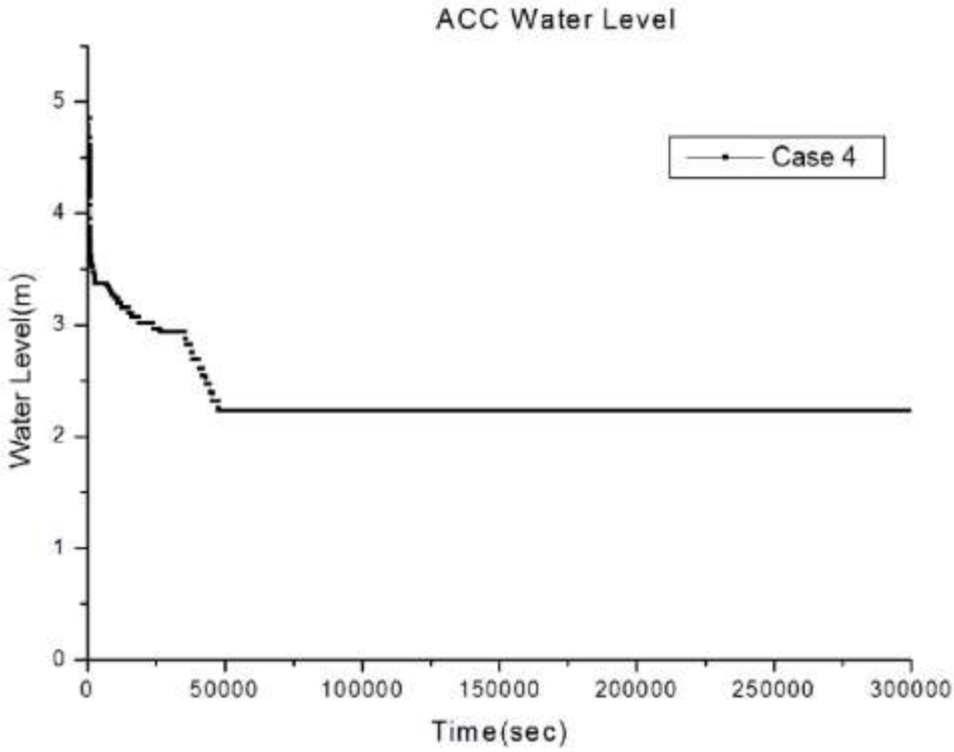

Figure 34 ACC Water Level of Case 4

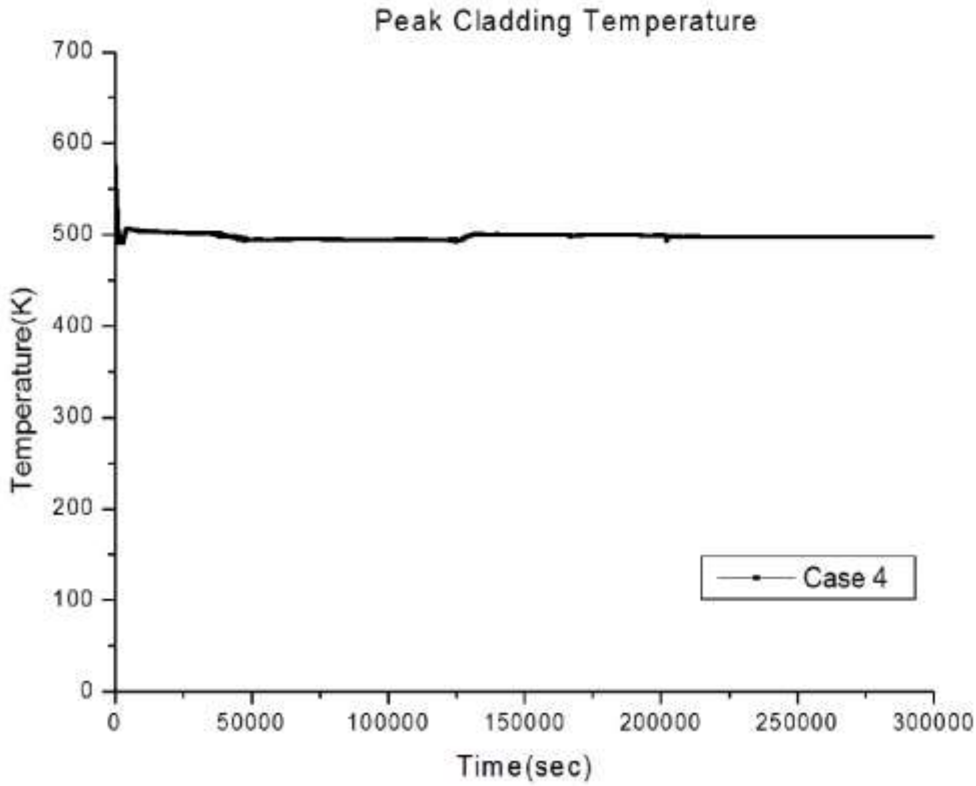

Figure 35 Peak Cladding Temperature of Case 4

According to the above results, it could be found that all 4 cases with mitigation strategy in this study could keep RCS water level above TAF, ensuring the safety function of reactor. Through the comparison between Case1~2 and Case 3~4, RCS could cooldown effectively during transient with the action of subsequent depressurization in URG strategy. And the time for URG cases which reached the full RCS water level is earlier than FLEX cases. Additionally, the URG could reduce the seal leakage rate effectively.

5 CONCLUSIONS

By using TRACE and SNAP codes, this study has developed a method to simulate and analysis the ELAP event for Maanshan NPP. Several conclusions are as follows:

- Once ELAP event occurred, there were about 60 hours to prepare multiple means of power and water supply to keep Maanshan NPP in a safe condition.
- The action of two steps depressurization for the URG could reduce the seal leakage rate effectively but still kept RCS water level above TAF.
- Two-step depressurizations could extend the time available to cope with the lineup of the alternate water.
- The URG or FLEX can keep RCS water level above TAF for the ELAP event.
- These results can help the decision making of mitigation strategy for the ELAP accident.

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NRC FORM 335
(12-2010)
NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/IA-0496

2. TITLE AND SUBTITLE

The Analysis and Study of ELAP Event and Mitigation Strategies Using TRACE Code for Maanshan PWR

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	**2019**

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Jung-Hua Yang, Jong-Rong Wang, Chunkuan Shish, Yu Chiang, Chun-Fu Huang, Shao-Wen Chen, Ting-Yi Wang*, Tzu-Yao Yu*

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Nuclear and New Energy Education and Research Foundation
Institute of Nuclear Engineering and Science,
National Tsing Hua University
101 Section 2, Kuang Fu Rd., HsinChu, Taiwan

*Department of Nuclear Safety
Taiwan Power Company
242,Section 3, Roosevelt Rd.,
Zhongzhena Dist.

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

In this study, TRACE code was used to evaluate the postulated Extended Loss of AC Power (ELAP) accident IN Maanshan nuclear power plant (NPP), determining whether RCS water level will drop down below Top of Active Fuel (TAF) while the 5^{th} diesel generator and gas turbines are all disabled when the accident occurred. The scenario and assumptions of postulated ELAP event in the study were referred to the WCAP-17601-P report and NUREG-1953 report. To analyze the effectiveness of URG and FLEX strategies, this research will run a base case without any mitigation strategy and four cases with multiple mitigation strategy under different conditions. According to the results of simulation, it can be found that all four cases in this study can keep RCS water level above TAF, ensuring the safety function of reactor.

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

Ultimate Response Guideline (URG)
Beyond-Design Base External Event (BDBEE)
Extended Loss-of-AC Power (ELAP)
Flexible and Diverse Coping Strategy (FLEX)
Top of Active Fuel (TAF)
Maanshan Nuclear Power Plant (PWR)

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)

unclassified

(This Report)

unclassified

15. NUMBER OF PAGES

16. PRICE

Printed on recycled paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMSSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS

@NRCgov

NUREG/IA-0496

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February 2019